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MASTER

1000 MWe

L M F B R

Liquid Metal Fast Breeder Reactor

FOLLOW-ON STUDY

WESTINGHOUSE ELECTRIC
TASK IV REPORT

Research and Development

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Prepared for ARGONNE NATIONAL LABORATORY

United States Atomic Energy Commission Division of Technical Information

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1000 MWe LMFBR FOLLOW-ON STUDY

TASK IV FINAL REPORT

March 1969

Prepared by the Advanced Reactors Division Staff

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SECTION I

INTRODUCTION

1. OBJECTIVE

The objective of the Follow-on Program as stated by the contract is:

To provide information which will be of assistance to the AEC in the furtherance of the National LMFBR Program Plan.

The objective of the current studies on the Nuclear Steam Supply System and Energy Conversion System was more precisely defined as:

formulation of a proposed Research and Development Program, as well as an appropriate assessment of other factors such as safety and conservation of national resources.

The above statements focus the intent of the Atomic Energy Commission sponsorship of these studies upon the importance of application of the results of concept formulation, conceptual design, and parametric and trade-off studies (Tasks I, II, and III) to the formulation of a course of action to achieve the goal of a safe, reliable, and economically viable LMFBR system.

2. SCOPE

The scope of a task to meet the above objectives is succinctly stated by the contract in establishing Task IV.

Concurrent with the preparation of the Final Report, the Contractor shall prepare a document outlining the required new data and information required for successful design and construction of the Contractor's reference design for the 1000 MWe LMFBR. Where sufficient uncertainties exist in the degree of attainment of these data and information, the effects on performance and power costs will be quantitatively reported using the results of the Task III parametric studies. In addition, an R and D program for obtaining these data and information will be supplied.

To assure that the data prepared under this task by all Follow-on Study contractors includes all the information required by the Atomic Energy Commission, the Argonne National Laboratory detailed the information required under each of the ten technical categories into which the LMFBR Program Plan is divided. That itemization is quoted below:

- a. A concise definition of the item or technical area requiring research and developing, sufficiently detailed to indicate clearly the bounds of the item or area, and the functional requirements where applicable
- b. A brief description of the base of current technology for the item, considered by the Contractor to be the starting point for new information needed; requirement to be satisfied by reference to "state-of-the-art" descriptions in the LMFBR Program Plan, where available and coinciding with the Contractor's opinions and understanding
- c. A clear statement of the new information needed under each item, including ranges of parameters, sizes, as appropriate, over which information is needed
- d. A program description for the item, including:
 1. Objectives, scope, technical justification and relationship to current programs
 2. Facilities and equipment required (existing adequate facilities indicated by reference only; new or modified facilities described)
 3. Manpower requirements
 4. Costs itemized for facilities, equipment and operating cost including manpower
 5. Time schedule, including identification and scheduling of major milestones

(In the event that an item is already covered adequately in a current on-going research and development program, the program description for this item need

consist only of a reference to the on-going program. If the on-going program will not produce the information the Contractor deems necessary, then the changes or extension necessary to satisfy the Contractor's requirements should be discussed.)

- e. An estimate of the probability of accomplishing the research and development objectives for the item within the funding and schedule specified
- f. A quantitative assessment of the effect on the reference concept design, and an estimate of the incremental penalty in the cost of energy produced by the reference concept power plant, which would result from nonattainment or only partial attainment of the research and development objective for the item; Contractor may indicate alternate paths to satisfy an objective, if desired.

3. APPROACH

3.1 Determination of Information Needs

To implement the task in such a way as to insure development of the data required in a form which could be readily related to the reference and alternate design descriptions, the following approach was used.

The system descriptions were systematically reviewed to identify all items or technical areas requiring research and development.

The item or area was identified through reference to the system description in the title of a work sheet denoting system and subsystem or component by abbreviated title as well as by a numerical index to the system description. Through this cross-reference, precise definition of bounds of the item or area may be obtained without duplication of all technical data.

Critical reference plant functional requirements were identified and listed in the first column of the work sheet. This listing does not include all functional requirements listed in the systems description, but only those which most significantly effect research and development needs. Entries in Column one (with cross-reference to the system's descriptions) provide the data required by item a. above.

In Column two, the base of current technology is summarized, providing as much parallelism with Reference Plant requirements as possible. Reference to the appropriate LMFBR Draft Program Plan "state-of-the-art" description is made. Reference to the Program Plan in Column two indicates general agreement with that description and further comment is made only if the Westinghouse assessment differs on a substantial point. The entries in Column two thus provide the data required by item b. above.

By comparison of Reference Plant requirements (Column one) with current technology, new information needed to implement the Reference Plant is identified and tabulated in Column three. These needs are the justification for research and development. The information needs are categorized by basic data, engineering data and techniques, experimental verification, and operational analysis. The needs identified in the first two categories in general may best fit in Base Technology Program with broad applicability. The needs categorized as experimental verification may be performed either under Base Technology Programs or Development Programs supporting a Demonstration Plant Project. Operational analysis will come primarily through Demonstration Plant Programs, the FFTF Project, and operation of major development facilities.

3.2 Program Plan

Section II of this report consists of the tabulation described above.

Section III of this report describes the program required to develop the new information needed to implement the design, and includes the information required in item d., with the exception of cost and manpower estimates which are provided in Section IV.

3.3 Value Analysis

Section V, Value Analysis, contains the results of the assessment required by item f. To establish the program for which this assessment would be made, judgments of the probability of accomplishing the research and development objectives for each item were made (item e.) so that emphasis could be placed

on those items with lesser probabilities of achievement, as well as those which are expected to be of the most significance economically.

3.4 Relation to LMFBR Program Plan

The close correspondence between the system description structure and the corresponding information needs and programs was purposely maintained to assure complete coverage, simplify cross-reference, and eliminate the need for the duplication of information. However, this does not permit compartmentalization of the data into the ten categories of the LMFBR Program Plans.

A cross-index is included to facilitate cross-referencing the tasks of Section III with appropriate volumes of the LMFBR Program Plan.

SECTION II

INFORMATION NEEDS

This section of the report describes the information and experience which will be needed to design, build and operate the 1000 MWe LMFBR.

These information needs were identified by reviewing the plant system descriptions contained in WARD-97, "1000 MWe LMFBR Follow-on Study, Tasks II and III Conceptual Design Report". Each table in this section, therefore, represents the required information for a particular subsystem or component as described in WARD-97. The system and subsystem or component from WARD-97 and the corresponding section and paragraph in that report are shown in the title box on each table.

Column one of the tables in this section contains the critical functional requirements of the subsystem or component in the 1000 MWe plant as abstracted from the appropriate paragraphs in WARD-97.

Column two includes a brief status of the current technology for the subsystem or component.

Column three lists the information needs required to implement the 1000 MWe Reference Plant obtained from the review discussed above.

Column four lists the R&D Task numbers from Section III of this report which are recommended in order to obtain the particular information or experience.

The titles of the tables in this section are shown in the following list.

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CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - (U, Pu)C Fuel (Paragraph 3.2-2)			
<p>Heat rating in excess of maximum 40 KW/ft required.</p> <p>Burnup in excess of 100 GWD/T average will be needed.</p> <p>Fuel shall be mixed carbide clad in Type 316 SS. Reference fuel - Cr-modified (U,Pu)C [modification will be made by addition $Cr_{23}C_6$ to hypo-stoichiometric (U,Pu)C].</p> <p>Alternate fuels - hyper-stoichiometric (U,Pu)C, stoichiometric (U,Pu)C, and oxycarbides (oxygen content approximately 1 wt%).</p> <p>Fuel form - solid cylindrical pellets (approximately 0.246 inches in diameter with a density of $90 \pm 5\%$ T.D.) sodium bonded to 316 SS cladding.</p> <p>Fuel will contain natural U and Pu with the Pu/(U + Pu) ratio ranging from 0.15 to 0.25.</p> <p>Reference fuel operating conditions - maximum fuel center temperature will be 1450°C, with a maximum burnup of 170,000 MWD/MT. Operating conditions for the alternate fuels have not been determined.</p> <p>Fuel will be compatible with the bond and cladding and will maintain sufficient structural integrity without crumbling or melting at normal and hot spot conditions to peak burnups of 170,000 MWD/MT.</p>	<p><u>Physical Properties</u></p> <p>Thermal conductivity - low temperature measurements (200 to 1000°C) on unirradiated (U,Pu)C have been made to determine the effect of Pu and carbon contents. No measurements have been made on irradiated material.</p> <p>Melting point - isolated measurements of the melting points of UC and (U,Pu)C have been made on unirradiated samples. Included in these works was preliminary determinations on the effects of Cr additions.</p> <p>Thermal expansion - measurements on a few samples with Pu/(Pu + U) ratios ranging from .05 to .2 have been made from 25 to 1400°C. These samples consisted of unmodified hypo-stoichiometric and stoichiometric fuel.</p> <p>Specific heat - no data is available on (U,Pu)C fuel.</p> <p>Vapor pressure - only a few conflicting measurements are available on mixed carbide material.</p> <p><u>Mechanical Properties</u></p> <p>Very little data are available on the mechanical properties of mixed carbide material. Some significant work has been conducted on the mechanical properties of UC.</p> <p>(U,Pu)C has been operated at the following ratings:</p> <p>28 KW/ft for gas bonded fuel in EBR-II.</p> <p>21 KW/ft for Na bonded fuel in GETR to 6.5 GWD/T</p>	<p><u>Engineering Data and Techniques</u></p> <p><u>Physical Properties</u></p> <p>Data on the physical properties of the unirradiated fuels are required for the reference fuel material and alternates. The effects of variations such as stoichiometry, amount and distribution of second phases, and the amount and distribution of porosity requires evaluation to optimize the fuel specification. Supplementing this unirradiated fuel characterization program will be determination of the changes in fuel properties as a function of fuel burnup.</p> <p>Specific data requirements in addition to the above general requirements:</p> <p>Thermal conductivity - measurements on the effect of sodium impregnation for the temperature range 500 - 1500°C.</p> <p>Bulk melting point and the second phase melting characterization.</p> <p>Specific heat up to the bulk melting point and latent heat of fusion.</p> <p>Vapor pressure data and vapor species identification.</p> <p><u>Mechanical Properties</u></p> <p>Testing will be required to determine appropriate elastic, plastic, and creep properties as functions of temperature and burnup.</p> <p><u>Compatibility Tests</u></p> <p>In- and out-of-pile tests to provide information on a wide range of variables at high temperatures (up to 1000°C), and establish the controlling factors on any reactions that may occur.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Variation of thermal conductivity of (U,Pu)C with U/Pu ratio for Pu/(U + Pu) range from 0% to 30% in the temperature range up to 2500°C.</p> <p>Samples of (U,Pu)C irradiated to burnups in excess of 170 GWD/T in EBR-II, FFTF or the Demonstration Plant to obtain data on swelling and fission gas release, migration of fission products, Pu and voids.</p>	<p>3.1.B</p> <p>3.1.B</p> <p>3.1.B</p> <p>3.1.D</p> <p>3.1.F</p> <p>3.1.G</p> <p>3.1.H</p> <p>3.1.B</p> <p>3.1.D, 3.1.F, 3.1.G, 3.1.H, 3.1.I, 7.3.A</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - (U, Pu) C Fuel (Paragraph 3.2-2) (Continued)			
<p>Fuel will have those physical and mechanical properties that make it capable of operating at high power ratings to peak burnups of 170,000 MWD/MT and still be within the reference design criteria for fuel swelling.</p> <p>Fuel swelling for the reference fuel pin as a function of temperature and burnup will cause a minimum of fuel-cladding interaction at the end of life.</p>	<p>Data on swelling and fission gas release is known for burnups up to 112 GWD/T only in a gas bonded fuel in a thermal flux in MTR. Na bonded fuel has been used in metallic form in EBR-II.</p>		

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Fuel Cladding (Paragraph 3.2-2)			
<p>Assurance of cladding integrity for burnups of 170 GWD/T is required with sufficient margin of safety to withstand 20% over-power conditions.</p> <p>The fission gas pressures exceed any encountered in the water reactor technology, and require either particularly strong cladding with a sizable fission gas plenum, or a vented design releasing the fission gases into the coolant stream.</p> <p>Cladding in thin-wall (~ 0.014 in.) tube of 316 stainless steel (~ 0.300-in. OD) in the fully annealed condition.</p> <p>Performance conditions:</p> <ol style="list-style-type: none"> Operating temperatures, 750-1400°C (400-750). Fast neutron flux, $1-5 \times 10^{23}$ n/cm². Sodium coolant velocity, up to 30 ft/sec. <p>Under these conditions, the cladding material must possess the following characteristics:</p> <ol style="list-style-type: none"> High yield strength. Low creep rate, long creep-rupture times and high stress for failure by low cycle fatigue. Sufficient metallurgical stability to prevent adverse effects on performance. 	<p>Experience with cladding integrity at high burnup is scanty, since the fast reactors operating presently, such as Dounraey, EBR-II and Fermi have metallic fuel which cannot sustain high burnups due to excessive swelling.</p> <p>Little experience is available for fuel rods with large gas plena or a vented design.</p> <p>Some data are available to support the combined performance requirements and conditions. Present programs are in existence to provide some design information.</p> <p>Bulk of available data is insufficient for establishing engineering design properties and allowances, although it does provide indication of performance capability to moderate temperature levels and neutron flux ($\sim 1300^\circ\text{F}$ (700°C) and 5×10^{21} nvt). Available physical property data do not account for potential changes due to radiation damage, i.e., void formation effects.</p> <p>State-of-the-Art reference:</p> <p>LMFBR Program Plan, Volume 7.</p> <p>WARD-33 Task I Final Report</p>	<p><u>Basic Data</u></p> <p>Mechanical properties: The following data are required in the uniaxial and biaxial stress conditions for unirradiated and irradiated material:</p> <ol style="list-style-type: none"> stress-strain curves creep and creep-rupture for continuously increasing stress system strain-cycle fatigue for range of $0-10^3$ cycles. <p>Metallurgical stability: The effects of variant degrees of cold work, grain size, structure and presence of all possible phases on the mechanical properties must be known.</p> <p><u>Engineering Data and Techniques</u></p> <p>Resistance to sodium corrosion: Expected material loss as a function of time with high temperature and flow velocities and without nuclear exposure.</p> <p>Resistance to radiation damage: In addition to the effects on mechanical properties and corrosion rates, complete understanding of the effects of void formation and growth on the performance of thin-wall tubes is required, including the effects of stress on void growth.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Fabrication and process control: A knowledge of the effects of various melting, fabrication, and forming processes on tube quality, including properties, and reproducibility to achieve a consistent and well-characterized tube via detailed specifications.</p> <p>With vented fuel rod design, the minimum cladding thickness which will sustain rigid fuel assembly structure and retain its integrity with 20% over power conditions should be determined.</p> <p>Fuel from the Demonstration Plant subjected to an extensive post-mortem analysis to determine the extent of embrittlement, erosion and the presence of failures.</p> <p>Carburization of cladding by C transfer from (U,Pu)C or carried by Na investigated in a reactor environment.</p> <p>The damaging effects of grid fretting on cladding determined and the preferred design tested in the Demonstration Plant.</p>	<p>3.1.C 3.1.F 3.1.1.a 3.1.G</p> <p>3.1.C 3.1.I 3.1.F 3.1.G</p> <p>3.1.C 3.1.I 4.3.A</p> <p>3.1.F 3.1.G 3.1.I</p> <p>3.1.C</p> <p>4.3.B 4.3.C</p> <p>3.1.F</p> <p>3.1.D</p> <p>4.3.C</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Fuel Cladding (Paragraph 3.2-2) (Continued)			
<p>d. Adequate resistance to radiation damage, i.e., sufficient ductility to accommodate mechanical and thermal strains and acceptable swelling rates due to void formation and growth.</p> <p>e. Low corrosion rate in sodium.</p> <p>f. Compatibility with (U,Pu)C in the presence of sodium.</p> <p>g. Sufficiently controlled fabrication processing to provide a consistent, high quality, low cost product.</p>			

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Fuel Bond (Sodium) (Paragraph 3.2-2)			
<p>The effective operation of Na bond between the ceramic fuel pellet and a metallic cladding requires that:</p> <p>a. no pockets of gas be formed along the length of the fuel resulting in the boiling of the sodium bond or fuel melting.</p> <p>b. the surfaces of SS 316 cladding bond and (U,Pu)C pellets should form a good continuous heat transfer system.</p>	<p>Thermodynamics and transport properties of Na are known up to 1900°C for Na of high purity.</p> <p>No measurements above these temperatures and for Na contaminated with fission products are currently available.</p> <p>There are indications that the impurities in Na can change the transport properties significantly.</p> <p>Thermodynamic properties available are published in three different systems of units requiring continuous reversion.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Studies of pressure surges caused by a nuclear transient and originating at the center of a gap between fuel pellets and propagating throughout the Na bond are needed to determine the extent of bond expulsion and the probability of cladding rupture.</p> <p>TREAT tests to determine the thresholds for cladding rupture and fuel meltdown for Na bonded (U,Pu)C fuel.</p> <p>Thermodynamic and transport properties of Na up to 2400°C (melting temperature of UC) and preferably up to 2800°C (melting temperature of UO₂).</p> <p>Tables of the thermodynamic and transport properties of Na prepared and suitable computer subroutines written in two versions: the British units and the MKS units to facilitate their inclusion into a nuclear - thermal-hydraulic computer code.</p> <p>The effect of the impurities in Na on all the transport properties of Na.</p> <p>The effect of pressure on the properties of Na vapor (equilibrium constants of isomeric structure) for temperatures and pressures up to the maximum anticipated for accidents conditions.</p> <p>The effect of impurities in Na on wetting of heat transfer surfaces and the thermal conductivity of Na bond.</p> <p><u>Environmental Verification and Operational Analysis</u></p> <p>The behavior of Na bonding in the presence of fuel pellets investigated to obtain the effect of Na penetration into the gaps between the pellets, into cracks in the pellets and into porosities; in and out of pile tests should be done.</p> <p>The effectiveness of the fuel rod design vented for the release of fission gases tested in the reactor environment of the Demonstration Plant as well as in test loops.</p> <p>The possibility of the formation of the axial void in (U,Pu)C under the action of a thermal gradient and high linear powers investigated experimentally. Should the phenomenon occur, a suitable analytical model will need to be formulated</p>	<p>4.2.A</p> <p>3.1.H</p> <p>4.2.B</p> <p>4.2.C</p> <p>4.2.D</p> <p>4.2.E</p> <p>4.2.F</p> <p>3.1.D</p> <p>3.1.E</p> <p>3.1.F</p> <p>3.1.G</p> <p>3.1.F</p> <p>3.1.G</p> <p>3.1.B</p> <p>3.1.F</p> <p>3.1.G</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Fuel Rod (Integral Performance)(Paragraph 3.2-2)			
<p><u>Fuel Rod Geometry</u></p> <p>Solid cylindrical (U,Pu)C fuel pellets.</p> <p>Pellets form a free standing stack.</p> <p>Axial blankets of UC pellets stacked above and below the fuel pellets.</p> <p>Fuel and blanket stacks are sodium bonded to the cladding.</p> <p>Cladding is formed from drawn type 316 stainless steel.</p> <p>The rod is sealed by fusion-welding in stainless steel end plugs.</p> <p><u>Fuel Rod Operating Conditions</u></p> <p>Fuel rods will receive burnups of 6×10^4 - 1.7×10^5 (average 10^5) MWD/T.</p> <p>Fuel temperatures will range from 450 - 760°C (surface) and 550 - 1450°C (center).</p> <p>The linear heat rating of the rod will be in the range 6 - 30 kw/ft (13.2 kw/ft average).</p> <p>Mean clad temperatures will be in the range 425 - 725°C.</p> <p>Fuel swelling will reduce the fuel-clad radial gap to a minimum of 0.5 mils at the end of life, i.e., fuel swelling will not impose strain on the cladding.</p> <p>Fission gases released from the fuel will be accommodated in a plenum above the upper axial blanket.</p>	<p>Small experimental batches of Cr-modified (U,Pu)C pellets have been manufactured and assembled in sodium-bonded fuel rods.</p> <p>Gas-bonded (U,Pu)C (unmodified) has been irradiated in a thermal flux to 100,000 MWD/T at 15-20 kw/ft (maximum fuel temperatures of 1150-1200°C).</p> <p>Sodium-bonded UC pins have been irradiated in a thermal flux to about 30,000 MWD/T.</p> <p>Sodium-bonded modified (U,Pu)C are currently being irradiated in a thermal flux. An interim examination at 30,000 MWD/T indicated satisfactory behavior.</p> <p>A lead fast-flux experiment on gas-bonded (U,Pu)C confirmed the thermal flux behavior at low burnup. Testing is continuing to high burnups.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Manufacturing Development</p> <p>Large scale fuel production processes (e.g., carbo-thermic reduction of oxides).</p> <p>Sodium-bonding techniques for long columns.</p> <p>Sodium bond inspection procedures.</p> <p>Fuel column centering system.</p> <p>Rod irradiation tests are required to provide the following data for candidate fuel-cladding combinations:</p> <p>Fuel swelling as a function of temperature, rating and burnup.</p> <p>Fission gas release as a function of temperature, rating and burnup.</p> <p>Effects of variations in fuel composition and constitution (amount and distribution of second phases) on swelling and fission gas release characteristics.</p> <p>The consequences of cladding rupture and partial sodium bond loss under normal operating conditions.</p> <p>The behavior of released fission gas bubbles in the sodium bond, particularly late in life when the bond width is small and the gas release rate is high.</p> <p>Small rod irradiation tests to provide data on:</p> <p>Fuel-sodium-cladding compatibility behavior.</p> <p>Buildup of fission product contamination in the sodium bond.</p> <p>Fuel temperature history as influenced by bond contamination, sodium impregnation and fuel density decrease.</p> <p>The effects of limited restraint by the cladding near to the end of life.</p>	<p>3.1.A</p> <p>3.1.E</p> <p>3.1.F</p> <p>3.1.G</p> <p>3.1.B</p> <p>3.1.F</p> <p>3.1.F 3.1.G 7.3.A</p> <p>3.1.B</p> <p>3.1.H</p> <p>3.1.F</p> <p>3.1.G</p> <p>3.1.F</p> <p>3.1.G</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Fuel Rod (Transient Behavior) (Paragraph 3.2-2)			
<p>Under foreseeable overpower conditions (~20% excess) there will be no gross fuel rod failure which could be propagated to adjacent pins.</p>	<p>Two tests (1 failure, 1 survival) have been made on unirradiated sodium-bonded UC.</p> <p>There have been no transient performance tests conducted on (U,Pu)C fuel. Some tests on individual rods have been planned by Westinghouse.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Conduct a transient test program to:</p> <p>Establish the limiting transient conditions that a rod will survive without gross failure.</p> <p>Investigate the mechanisms and kinetics of rod failure.</p>	<p>3.1.H</p> <p>3.1.H</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Fuel Rod Spacer Grids (Paragraph 3.2-3)			
<p>Hold fuel rod in position and prevent vibration at temperatures up to 1200°F and coolant flow rates up to 30 feet/sec.</p> <p>Prevent thermal bowing.</p>	<p>Fermi Core A employed grids brazed with Coast Metals Alloy 60. After 30,000 hours operation* the grids were visually inspected and found to be in good condition.</p> <p>* Coolant flow rate - 15 feet/sec Thermal history - < 1 hr. at 1150 ° F Approximately 750 hr. > 550° F Remainder < 550° F</p>	<p><u>Engineering Data and Techniques</u></p> <p>Sodium corrosion resistance and mass transfer characteristic of candidate grid materials and brazing alloys.</p> <p>Effects of sodium environment on grid material mechanical properties.</p> <p>Effects of fabrication procedures on performance of grid jointures in sodium.</p> <p>Irradiation effects on candidate grid materials and brazing alloys.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Grid performance evaluation under reactor conditions as closely as possible simulating reference plant conditions.</p>	<p>3.2.A</p> <p>3.2.A</p> <p>3.2.A</p> <p>3.2.A</p> <p>3.2.A</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Fuel Assembly (Paragraph 3.2-3, 3.2-4)			
<p>Fuel assembly of a structure which:</p> <ol style="list-style-type: none"> Will operate with minimum vibration. Will minimize thermal distortion in the lateral direction. Will sustain coolant velocities equal to or better than 30 ft/sec. Will allow power densities of 487 kw/liter. Will withstand hot channel coolant temperatures of 1300°F. Will operate with axial support or hold down mechanism. 	<p>Metallic fuel rods assembled individually into a core were used in the EBR-I reactor, and fuel assemblies with an eggcrate grid in the Fermi reactor. Both cores were damaged through flow stoppages or disturbances.</p> <p>FFTF uses hydraulic hold-down and is contemplating mechanical pre-stressing to eliminate thermal bowing effects.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>In-reactor heat transfer tests carried out to ascertain the effects of complex thermal gradients and thermal bowing on the thermal performance and stability of the assemblies.</p> <p>Determine film coefficient correlations under steady state and transient conditions.</p> <p>The effects of large thermal gradients on fuel support system and the performance of strongback temperature compensators determined.</p> <p>Tests in water loops of partial and full assemblies to determine the hydraulic resistance under steady state and transient conditions and the vibration characteristics of the assembly.</p> <p>Experiments in over-sized water loops to ascertain the velocity profiles, flow distribution characteristic effects of coolant jetting and to identify the areas where cavitation might occur.</p> <p>Coolant inlet design selected to assure coolant flow-free from stoppages.</p> <p>Determine the radiation, temperature, stress effect on creep of structural materials, in particular Mo, SS 304, and SS 316.</p>	<p>4.5.A</p> <p>4.5.B</p> <p>3.2.A 4.5.C</p> <p>4.5.D</p> <p>4.5.E</p> <p>4.5.E</p> <p>3.2.A</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Fuel Temperature Coefficient (Subsection 3.5)			
<p>Total temperature coefficient non-positive and preferably negative.</p> <p>Sign and magnitude of Doppler effect in carbide fuel rods containing ~20% Pu in (U,Pu)C.</p> <p>Temperature effect of changes in bulk Na at temperatures ~523°C.</p> <p>Temperature effect on reactivity of core expansion at ~523°C.</p>	<p>Considerable doubt exists concerning the magnitude and sign of the temperature coefficient particularly its fissile Doppler component.</p> <p>Only two measurements of the Doppler effect in the ZPR assemblies are available so far.</p> <p>Measurements in EBR-II are for low Pu fuel of <6% pu.</p> <p>Measurements in Fermi are for metallic fuel.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Better methods of measuring the neutron spectrum in the resonance region, for a wide range of U,Pu/Na ratios in cores of composition typical for an LMFR.</p> <p>Better understanding of the effect of the U/Pu ratio on the magnitude of the Doppler coefficient, from measurements over the range of U/Pu compositions of interest up to 30% Pu/(U + Pu).</p> <p>Measurements of the Doppler effect for whole core regions, rather than single rods, to magnify the effect and resolve the doubts regarding its magnitude in critical experiments.</p> <p>Experiments designed to obtain better resolution between the reactivity effect of temperature expansion of the core, density change in Na and the Doppler effect.</p> <p>Doppler effect measurement in the SEFOR, FFTF and the Demonstration Plant to verify the values predicted on the basis of critical experiments.</p>	<p>4.1.A</p> <p>4.1.B</p> <p>4.1.C</p> <p>4.1.D</p> <p>4.1.E</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Na Coolant (Subsection 3.4)			
Prevention of the onset of local boiling of Na and, should it occur, prevention of large Na void formation and its propagation throughout the core.	<p>Failures due to Na coolant blockage or expulsion have occurred in:</p> <p>a. EBR-I reactor</p> <p>b. Fermi reactor</p> <p>Both failures occurred accidentally, core instrumentation was inadequate to predict the onset of failure or its location. Design of the core in a) was such as to lead to gross meltdown, and in b) limited the damage to two assemblies.</p> <p>Data on the onset of Na boiling are available for pool boiling of highly purified Na only.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Better understanding of the mechanism of the onset of Na boiling, with particular reference to the effects of impurities in Na, or the condition of heat transfer surfaces.</p> <p>Studies of advantages and disadvantages of canned fuel assembly for preventing void propagation throughout the core.</p> <p>Determination of the time-reactivity sequence of voiding transients and the selection of voiding criteria, including the maximum reactivity insertion which can be controlled automatically by negative temperature effects and mechanically by control rod insertion.</p> <p><u>Experimental Verification and Operational Analysis.</u></p> <p>Non-destructive critical experiments to confirm the space dependence of the sign of the void effect. Selection of optimum geometry for desired void effect.</p> <p>Tests of advantages of modular cores in reducing or localizing gross void formation, based on multi-modular criticals in ZPPR assembly.</p> <p>Studies of effects of local boiling and entrained gas in the Na coolant on the thermal performance of the fuel.</p> <p>Repeat experimental tests of SPERT type for nuclear transients to determine the conditions under which gross core voiding occurs, to select geometries which hamper void formation, and to measure the energy releases which take place.</p>	<p>4.4.A</p> <p>4.4.B</p> <p>4.4.C</p> <p>4.4.D</p> <p>4.4.E</p> <p>4.4.F</p> <p>4.4.G</p> <p>4.4.H</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Core Modules (Subsection 3.5)			
<p>Determine the magnitude of neutronic coupling between the core modules.</p> <p>Worth of control rods and Na voids in the adjacent modules.</p> <p>Time constants of the reactivity feedback effects in coupling between the modules.</p>	<p>No coupled fast reactors have been operated so far in a modular geometry.</p> <p>Some neutronic coupling data has been obtained for the parts of the seed assemblies separated by the fertile blanket in the Shippingport reactor. Applicability of this data to fast reactors is in doubt.</p> <p>Point kinetic models commonly used in describing the time behavior of reactors are totally inadequate for describing kinetic behavior of modular cores.</p> <p>Rover coupled core experiments involved very loosely coupled cores and not separated by a blanket.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Analytical work to produce space-time kinetic model applicable to the modular geometry.</p> <p>The analytical model tested against experiment in a modular geometry with at least 2 cores separated by a blanket, and preferably 4 cores, to correspond to the geometry proposed for the 1000 MWe LMFBR.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Critical experiments in modular geometries for the measurement of transfer functions, neutronic coupling between the modules.</p> <p>Control rod worths measured to determine rod interaction (if any) between rods in different modules. Similarly, the interactions of modules, when voiding of Na occurs in one of them, studied experimentally.</p>	<p>4.8.B</p> <p>4.8.A</p> <p>4.8.C</p> <p>4.8.D</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Appendix A) - Non-Modular Core (Appendix A)			
Safe operation combined with optimum economics.	Virtually all operating reactors today have non-modular cores. In large fast reactors in all geometries except extremely flat pancakes or very slender cylinders, the interior of the core, when voided, produces a positive reactivity effect. This has heretofore forced the size of the fast reactors to be kept small.	<u>Engineering Data and Techniques</u> The mechanism of void formation in the core investigated more thoroughly to determine the sequence of events in the formation of a large void in view of its importance in the non-modular cores. Study the probability of propagation of void throughout the non-modular core loaded with uncanned fuel assemblies. The extent of severe damage in the meltdown of the core and the energy releases involved determined for the modular and the non-modular concepts.	4.11.A 4.11.B 4.11.C

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Computer-Assisted Fuel Management System (Paragraph 3.5-2)			
<p>A system continuously assessing the isotopic inventory of the core, and keeping the record of the total energy generated by the core.</p> <p>The system should be capable of identifying the burnup condition of each fuel assembly, and its isotopic inventory, calculating current value of the breeding ratio, and indicating which assemblies should be refueled and which recycled.</p> <p>The system should be based on a sufficient number of in-core detectors, analogue-to-digital converter, storage unit, and a digital computer to carry out the recording-predicting function.</p>	<p>The following reactors are equipped with on-line computers:</p> <ul style="list-style-type: none"> a. French RAPSODIE b. British DOUNREAY c. Hinkley Point B <p>The computers digest information from thermocouples and flowmeters to obtain total energy releases and energy releases in each assembly. This is possible because of canned design of the fuel assembly. Information for kinetic control is also available.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Evaluation of the economic advantages of a continuously monitored fuel management scheme.</p> <p>Depending on the results of the above noted economic evaluation, the following research items (below) should be performed.</p> <p>Evaluation of the advantages of a canned fuel assembly versus an uncanned fuel assembly, from the point of view of being able to map radial power distribution, locate over-heating fuel assemblies, and record total energy released from each assembly.</p> <p>Development of in-core sensors for the measurement of axial power distributions and the selection of the flux wire, Aeroball or miniature fission chamber detectors for this function.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Development of non-destructive method to sense the changes in the U/Pu ratio in each assembly. Removable fuel sections or insertable devices containing fuel samples of proper U/Pu ratio should be developed to sense the changes in the inventory of heavy metal isotopes and feed them into the fuel management computer.</p> <p>The in-core fuel status sensing system should be installed in the Demonstration Plant and checked out against the results of the post-mortem isotopic analysis of the spent fuel.</p>	<p>4.10.A</p> <p>4.10.B</p> <p>4.10.C</p> <p>4.10.D</p> <p>4.10.E</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Radial Blanket Assembly (Paragraph 3.2-3,3.2-4)			
<p>Temperature gradients of the assemblies at the core-blanket interface are in the 150°C range. Inward bowing of the radial blanket must be minimized to avoid the increase in reactivity due to resulting core compaction.</p> <p>An effective orificing system is needed to allow efficient thermal utilization of the blanket, operating throughout its life under grossly shifting power distributions.</p>	<p>Fuel assemblies loaded with UO₂ fuel rods are used extensively in water reactors but subject to much lower temperature gradients, temperatures, flux magnitudes and lengths of irradiation.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Testing of various assembly configurations, particularly flow deflecting rod wrappers or baffles, to assure effective mixing of coolant in the canned radial blanket assembly.</p> <p>Orificing and flow diverting systems tested under reactor operating conditions in the Demonstration Plant to assure an efficient system is developed capable of handling the large power shifts in the radial blanket occurring as a result of Pu buildup.</p> <p>Testing of a rod support system which will contain fuel rod deflection and vibration within reasonable limits.</p> <p>Experimental testing to determine the temperature distribution in the rods and in the assembly to validate the design selected for the LMFBR.</p> <p>Experimental testing to determine the effect of the sodium environment and the fast flux environment existing in the blanket region on the blanket wrapper can.</p>	<p>4.6.A</p> <p>4.6.B</p> <p>4.6.C</p> <p>4.6.D</p> <p>3.2.B</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Vessel Internals (Paragraph 3.3-2, 3.4-2.5)			
<p>The structure must be free from vibration leading to core damage or flow maldistribution.</p> <p>30 years design life is required.</p> <p>Accessibility for maintenance and inspection is required.</p> <p>Assure appropriate Na velocity profile at core inlet.</p>	<p>Present technology is based on light water reactors and shows problems in the design for assuring structural integrity of core barrel and thermal shields due to flow induced vibration.</p> <p>Fermi reactor has had problems with breaking away of vessel internals and resultant flow blockage.</p>	<p><u>Engineering Data and Techniques</u></p> <p>The irradiation, temperature, stress and time effects on the structural materials and weldments determined.</p> <p>Demonstration Plant properly instrumented to verify the temperatures, displacements and vibration frequencies of the core internals.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Scale model of the core and internals tested to determine flow distribution and hydraulic resistance and vibration characteristics of the modular and alternate designs.</p> <p>Development and testing of an effective siphon breaker to prevent the draining of Na from the core.</p> <p>The effect of location of the vortex suppressor plate on pressure drop and flow distribution.</p>	<p>3.2.E 3.2.F</p> <p>4.9.C</p> <p>4.9.A 4.9.B 4.9.D</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Control Rods and Drives (Sub-section 3.6)			
<p>Regulate the neutron flux to provide desired rate of energy generation.</p> <p>Balance the power distribution in the core and provide operational stability.</p> <p>Compensate for reactivity changes caused by burnup, temperature, power, and loading conditions.</p> <p>Provide emergency control of reactivity.</p> <p>Provide a shutdown margin of safety during fuel handling or maintenance.</p> <p>Jam free operation of control rod drives is required, combined with long term freedom from deposition of fission products, Na freeze out or effects of mass transfer. The performance of control rod blades must not be impaired by distortion, attack by fission products or erosion by mass transfer in Na.</p> <p>The size of steps in control rod movement should be limited to less than 1.2 c/sec withdrawal and 1.5 sec for insertion time.</p> <p>The size of the insertion steps is limited by the small temperature coefficients of the core.</p> <p>The temperature gradient across the control rod must not be excessive.</p>	<p>Control rod material in Fermi is B^{10} contained in boron carbide.</p> <p>Control rods in EBR-II are similar to the standard fuel elements in all respects except size.</p> <p>SRE Core II control rod material was 2% boron-nickel alloy in the form of rings suspended on a pull tube.</p> <p>LMFBR Program Plan, Volume 6 and Volume 3.</p> <p>Magnetic jack mechanisms are used extensively in water reactors at present. Their performance in Na at high temperature and contaminated by fission products has not been proven as yet.</p> <p>Tests of this type of mechanism with modifications to reduce the size of step in rod movement are currently in progress at (W) for the FFTF reactor.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Effect of sodium and neutron flux environment on the material selected for the control assemblies, and the various fabrication methods used for assembly.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Coefficients of sliding friction determined for each combination of materials in relative motion in control rod guide tubes and control rod drives under condition of high temperature Na.</p> <p>Deposition of volatile fission products, present in Na with vented fuel design, in the control rod drive mechanisms and their effect on the smooth operation of the drives determined in an operating reactor.</p> <p>The necessity of internal cooling of the large control rods, needed to provide adequate control tested by measuring the amount of distortion in Ta rods with and without internal cooling channels.</p> <p>Reactivity worth of T_h control rods obtained in the Demonstration Plant at different burnups and radial positions.</p> <p>Changes in reactivity worth of control rods located at the core blanket interface and due to Pu buildup checked using the high burnup condition of the Demonstration Plant at the end of the refueling cycle.</p>	<p>3.2.D</p> <p>4.7.A</p> <p>4.7.B 7.3.C</p> <p>4.7.C</p> <p>4.7.D</p> <p>4.7.E</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR CORE (Section 3) - Spent Fuel Shipping Cask (Paragraph 3.2-4)			
<p>A shipping cask and carrier (either truck or railroad car) will be required to ship the spent blanket and core assemblies from the plant spent fuel cooling pit to the reprocessing site. Minimum cooling time is desired for economic purposes.</p>	<p>Shipping casks are available for thermal reactor plant spent fuel elements, however, these casks do not have the capability to discharge the heat generated in spent fast reactor elements for reasonable cooling periods, i.e., 3-4 months.</p> <p>LMFBR's will probably require a shipping cask containing separately canned assemblies. These steel cans could be filled with a liquid-metal to distribute the heat more evenly throughout the cask thereby enhancing its heat removal capabilities, while reducing the shielding weight. ORNL has found that high decay heats have forced them to go to a "six-gun" design—an all-steel cask approximating the shape of a revolver cylinder.</p>	<p><u>Engineering Data and Techniques</u></p> <p>The maximum allowable spent fuel shipping cost as a function of reactor output is needed. (This will dictate the maximum allowable cooling time and cask development work necessary.) 5.7.N</p> <p>Natural convection and radiation heat transfer rates from a liquid-metal filled can containing a spent assembly must be determined. 5.7.N</p> <p>Effect of natural circulation on cask heat dissipation must be determined. 5.7.N</p> <p>Information on head-end processes and the amount of fuel required in a batch at the reprocessing plant is needed. 5.7.N</p> <p>Data on fuel burnup; fuel maximum/average exposure; load factor; fuel assembly size, loading, and maximum allowable clad surface temperature; fuel value and use charge are required. 5.7.N 4.10.E</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>The integrity of the selected shipping cask must be determined for the maximum credible accident situation in 10 CFR-71. 5.7.N</p>	

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR VESSEL (Section 4) - Reactor Vessel (Paragraph 4-2.1)			
<p>Configuration: single wall, 17.25 ft. O.D., 45 ft. length, removable shield plug, supported at lateral fuel port, open head refueling with lateral fuel handling.</p> <p>Design Life: 30 yr.</p> <p>Code: ASME III and Code Case 1331</p> <p>Design Pressure: 200 psi</p> <p>Design Temperature: 850°F</p> <p>Operating Temperature: 770°F</p> <p>Wall Thickness: 2.25 and 4.25 inches</p> <p>NVT: 9.5×10^{19} (>.85 Mev)</p> <p>Material: 304 SS</p> <p>Transients:</p> <p>Outlet: ~37°F/sec for 4.5 sec., then 8.5°F/sec for 7 sec.</p> <p>Inlet: ~3°F/sec for 13 sec., then +4°F/sec for 13 sec.</p>	<p><u>Hallam</u></p> <p>Configuration: single wall, diameter 19 ft., 33 ft. length, supported on base, through plug refueling.</p> <p>Material: 304 SS</p> <p><u>SRE</u></p> <p>Configuration: single wall, diameter 11 ft., 19 ft. length, supported on base, through plug refueling.</p> <p>Design Pressure: 14 psi</p> <p>Design Temperature: 1472°F</p> <p>Operating Temperature: 1200°F</p> <p>Wall Thickness: 1-1/2 in.</p> <p>Material: 304 SS</p> <p><u>SEFOR</u></p> <p>Configuration: single wall, diameter 40 in. at core, 57 in. upper, 162 in. length, hung from upper portion of vessel, open head refueling.</p> <p>Code: ASME III and Code Case 1331</p> <p>Design Pressure: 65 psi</p> <p>Design Temperature: 1050°F</p> <p>Wall Thickness: 3/8 in. lower, 3/4 in. upper</p> <p>NVT: 7×10^{22} (>1 Mev)</p> <p>Materials: 304 SS</p> <p>State-of-the-Art reference:</p> <p>LMFBR Program Plan, Volume 3.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Design standards, rules and allowable stresses for high temperature design. Specific needs include a) biaxial creep, b) stress-rupture, and c) low-cycle fatigue.</p> <p>Development of analytical procedures. Specific needs include: a) evaluation of current techniques with recommendation, and b) development of new procedures.</p> <p>In-service inspection requirements. Specific needs include: a) techniques, b) procedures, and c) inspection frequency.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Reactor vessel closures and seals. Specific needs include the development and testing of hardware design to determine: a) remote operational abilities of closures, and b) remote operational abilities of handling and accurately positioning of large masses such as the shield plug. Both cases involve extreme environments. Also the testing of seals and sealing materials to determine environmental effects for long periods and testing of prototype shield plug cooling systems to provide dimensional and distortion stability.</p>	<p>5.1.A</p> <p>5.1.B</p> <p>5.1.C</p> <p>5.1.D</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PRIMARY AND INTERMEDIATE REACTOR COOLANT SYSTEMS (Sections 6 and 7) - Primary and Intermediate Sodium Pumps (Paragraphs 6-2.1.3 and 7-2.1.2)			
<p>Type: Mechanical, single stage, centrifugal with free sodium surface. A vertical shaft driven by a wound rotor induction motor with liquid rheostat speed control.</p> <p><u>Primary Pumps</u></p> <p>Capacity: 149,200 gpm</p> <p>Dynamic Head: 300 ft. of Na</p> <p>Operating Temperature: 770°F</p> <p>Design Temperature: 800°F</p> <p>Material: 304 SS</p> <p><u>Intermediate Pump</u></p> <p>Capacity: 103,500 gpm</p> <p>Dynamic Head: 140 ft. of Na</p> <p>Operating Temperature: 650°F</p> <p>Design Temperature: 700°F</p> <p>Material: 304 SS</p>	<p>The two pumps built to date which most closely resemble the reference design are the Fermi and Hallam pumps.</p> <p><u>Fermi</u></p> <p>Capacity: 11,800 gpm</p> <p>Dynamic Head: 310 ft. of Na</p> <p>Operating Temperature: 550°F</p> <p>Design Temperature: 1000°F</p> <p>Operating Time: Intermittent for 5 years up to 1000°F</p> <p>Material: 304 SS</p> <p><u>Hallam</u></p> <p>Capacity: 7200 gpm</p> <p>Dynamic Head: 160 ft. of Na</p> <p>Operating Temperature: 945°F</p> <p>Design Temperature: 100°F</p> <p>Operating Time: 800 hrs. at 900-1000°F</p> <p><u>EBR-II</u></p> <p>Operating Temperature: 700°F</p> <p>Design Temperature: 800°F</p> <p>Operating Time: "Many thousands of hours"</p> <p><u>S.R.E.</u></p> <p>Operating Temperature: 950°F</p> <p>Design Temperature: 1000°F</p> <p>Operating Time: 800 hrs.</p>	<p><u>Basic Data</u></p> <p>Mechanical property data for 304 SS in sodium for the temperature range 700-800°F.</p> <p>a) Tensile Data (U.T.S. Yield Ductility)</p> <p>b) Fatigue Data (High Cycle)</p> <p>c) Creep and Stress Rupture Data</p> <p>Identification of suitable bearing materials, on the basis of wear in sodium, thermal properties (conductivity, expansion), sodium corrosion and mass transfer, under start up and low head conditions.</p> <p>Corrosion-erosion data for pump impeller materials.</p> <p><u>Engineering Data and Techniques</u></p> <p>Hydraulic data on specific pump concepts in order to optimize the pump hydraulic performance, and to assure good agreement between predicted and actual performance, including variation of head, flow and NPSH requirements with varying sizes and designs of impellers, approach channels, guide vanes, etc.</p> <p>Data concerning the overall behavior of and resulting effects on pump operation of the sodium free surface within the pump tank, including ability to control sodium level, potential for gas entrainment, etc.</p> <p>Data on the resulting stress and/or deformation in critical areas (areas where close tolerance must be maintained) due to the severe thermal environment.</p> <p>Data concerning the dynamic characteristics of all rotating elements and its relation to vibration of these elements and the pump body.</p> <p>Mechanical design bases for specific pump sub-components. Included are static seals, and considerations for assembly, disassembly, and maintenance of the unit.</p> <p>Fabrication and techniques for large bearings, impeller shafts, and casings.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>A gas type shaft seal for shaft diameters of approximately 8-12 inches. Specific needs include: a) suitable seal materials, b) formulation of acceptable seal performance characteristics, and c) seal proof tests in specially designed rigs and prototype pumps.</p>	<p>3.5.A</p> <p>3.5.B</p> <p>5.4.B</p> <p>3.5.D</p> <p>5.4.C</p> <p>5.4.C</p> <p>5.4.D</p> <p>5.4.E</p> <p>5.4.F</p> <p>3.5.C</p> <p>5.4.A</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PRIMARY AND INTERMEDIATE REACTOR COOLANT SYSTEMS (Section 6 and 7) - Primary and Intermediate Sodium Pumps (Paragraphs 6-2.1.3 and 7-2.1.2) (Continued)			
	<p>Some corrosion and mechanical properties in Na data are available for austenitic materials, but not under operating conditions.</p> <p>State-of-the-Art reference:</p> <p>LMFBR Program Plan, Volume 3.</p>	<p>A radial bearing compatible with 8-12 inch shaft diameters. Specific needs include: a) determining actual bearing loads that will be encountered, b) bearing materials suited to this application, and c) proof tests of bearing designs in specially designed rigs and prototype pumps.</p> <p>Proof tests of and operational analysis of prototype pumps to verify design and provide a base for further extrapolation.</p> <p>Fabrication techniques for large bearings, impellers, shafts and casing need to be resolved.</p>	<p>5.4.B</p> <p>5.4.G</p> <p>3.5.C</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PRIMARY REACTOR COOLANT SYSTEM (Section 6) - Intermediate Heat Exchanger (Sodium to Sodium)(Paragraph 6-2.1.2)			
<p><u>General</u></p> <p>Type: vertical, counter flow, single-wall tube, shell-and-tube design</p> <p>Thermal Rating: 1300 MWt</p> <p><u>Shellside</u></p> <p>Fluid: radioactive sodium</p> <p>Flow: 63.7×10^6 lb/hr</p> <p>Inlet Temperature: 1000°F</p> <p>Outlet Temperature: 770°F</p> <p>Material: 304 SS</p> <p><u>Tubeside</u></p> <p>Fluid: non-radioactive sodium (secondary sodium)</p> <p>Flow: 47.2×10^6 lb/hr</p> <p>Inlet Temperature: 647°F</p> <p>Outlet Temperature: 970°F</p> <p>Material: 304 SS</p>	<p>Sodium corrosion - very little available on Type 304 SS.</p> <p>Mechanical properties in sodium - data available on Type 304 SS mostly obtained under conditions not representative of those expected in service.</p> <p>Intermediate Heat Exchangers already in use, which are similar in design and materials to present system, are at Fermi and EBR-II.</p> <p><u>Fermi</u></p> <p><u>General</u></p> <p>Type: vertical, single-wall tube, shell-and-tube design</p> <p>Thermal Rating: 143 MWt</p> <p><u>Shellside</u></p> <p>Fluid: radioactive sodium</p> <p>Flow: 5.3×10^6 lb/hr</p> <p>Inlet Temperature: 900°F</p> <p>Outlet Temperature: 600°F</p> <p><u>Tubeside</u></p> <p>Fluid: non-radioactive sodium</p> <p>Flow Rate: 5.3×10^6 lb/hr</p> <p>Inlet Temperature: 520°F</p> <p>Outlet Temperature: 820°F</p> <p><u>EBR-II</u></p> <p><u>Shellside</u></p> <p>Inlet Temperature: 888°F</p> <p>Outlet Temperature: 695°F</p> <p><u>Tubeside</u></p> <p>Inlet Temperature: 588°F</p> <p>Outlet Temperature: 866°F</p> <p>(Has not been subjected to design temperature.)</p> <p>State-of-the-Art reference:</p> <p>LMFBR Program Plan, Volume 3.</p>	<p><u>Basic Data</u></p> <p>Sodium corrosion and erosion data on Type 304 SS under representative service conditions, specifically in deposition areas.</p> <p>Data on mechanical properties of Type 304 SS in sodium under representative service conditions. Main emphasis to be placed on creep, stress-rupture, stress relaxation and fatigue.</p> <p>Effect of fission products on the structure and mechanical properties of IHX materials.</p> <p><u>Engineering Data and Techniques</u></p> <p>Design criteria based on a) the current state-of-the-art, and b) a review and assessment of the operational aspects of liquid-metal heat transfer components. This criteria must provide standards and guidelines for IHX design (mechanical and functional), operation, manufacture, installation, maintenance, and component quality (i.e., inspection and quality assurance).</p> <p>Information, both analytical and experimental, to permit accurate thermal, hydraulic, and mechanical analyses. Information is principally needed for the verification, refinement and/or development of techniques to determine flow stability, flow distribution, vibration, and transient temperature distributions in sodium-to-sodium heat exchangers.</p> <p>A fabrication and quality control technology consistent with the safety and reliability demands established by design criteria. Developed technology must provide a) successful methods for tube-to-tubesheet joint attachment, b) proven techniques for the welding and fabrication of reference materials, and c) for the assurance of component quality, such as techniques and procedures for in-service and shop inspection, and the documentation of inspection data.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Prooftesting of prototype heat exchangers to verify analytical performance predictions and to demonstrate mechanical reliability.</p>	<p>3.4.A</p> <p>3.4.B</p> <p>3.4.C</p> <p>5.5.C</p> <p>5.5.D</p> <p>5.5.F</p> <p>5.5.A</p> <p>5.5.B</p> <p>5.5.E</p> <p>5.5.F</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PRIMARY REACTOR COOLANT SYSTEM (Section 6) - Check Valves (Paragraph 6-2.1.3)			
<p>Type: To be determined</p> <p>Size: 48-50" diameter</p> <p>Operating Temperature: 770°F</p> <p>Design Temperature: 800°F</p> <p>Operating Flow Rate: 63.7 x 10⁶ lb/hr</p> <p>Transients:</p> <p> ~3°F/sec for 13 sec. then +4°F/sec for 13 sec.</p> <p>Note: A check valve is also part of the Auxiliary Cooling System (Section 9), having a 22" diameter for 700°F operation (Paragraph 9-2.1.3). This valve design is assumed to be less critical than the one described above, and thus, to be inherently covered in the corresponding R and D programs.</p>	<p>The two largest operational sodium system check valves are:</p> <p><u>Ferri</u></p> <p>Type: Balanced disc</p> <p>Size: 16 inch diameter</p> <p>Operating Temperature: 550°F</p> <p>Operating Flow Rate: 2.95 x 10⁶ lb/hr</p> <p>Material: 304 SS</p> <p><u>Hallam</u></p> <p>Type: Swing Gate</p> <p>Size: 14 inch diameter</p> <p>Operating Temperature: 550-950°F</p> <p>Operating Flow Rate: 2.80 x 10⁶ lb/hr</p> <p>Material: 304 SS</p> <p>State-of-the-art Reference:</p> <p> LMFBR Program Plan, Volume 3.</p>	<p><u>Basic Data</u></p> <p>Identify materials for:</p> <ol style="list-style-type: none"> Structural components Hard facing for valve seats Spring materials <p>Mechanical properties in Na for the temperature 700-800°F:</p> <ol style="list-style-type: none"> Tensile (UTS Yield Ductility) Fatigue Data (High Cycle) Creep and stress rupture data for 30 yr. life <p>Testing of hard face materials under sliding wear conditions in sodium.</p> <p><u>Engineering Data and Techniques</u></p> <p>Development to determine design concepts for check valves up to 50" diameter.</p> <p>Development to determine and minimize thermal stress and distortion effect relative to specific check valve designs.</p> <p>Development to determine sodium oxide deposit characteristic on check valve sub-assemblies and a design to</p> <ol style="list-style-type: none"> prevent oxides deposits or with features to break oxides deposit and/or self-welding bond. <p>Development to determine the operating characteristics of check valve designs to insure sodium hammer will not occur when valve closes.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Proof tests and operational analysis of prototype check valve designs.</p> <p>Fabrication: Application of hard facing.</p>	<p>3.6.A</p> <p>3.6.B</p> <p>3.6.C</p> <p>5.3.A</p> <p>5.3.B</p> <p>5.3.C</p> <p>5.3.D</p> <p>5.3.E</p> <p>3.6.D</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PRIMARY REACTOR COOLANT SYSTEM (Section 6) - Main Coolant Piping (Paragraph 6-2.1.5)			
<p>Type: Single-walled, fusion welded, stainless steel pipe.</p> <p>Thermal expansion accommodated by expansion loops throughout system.</p> <p>Size: 48 in. O.D.</p> <p>Operating Temperature: 1000°F</p> <p>Transients:</p> <p>Hot Leg: ~-37°F/sec for 4.5 sec., then -4.5°F/sec for 7 sec.</p> <p>Cold Leg: ~-3°F/sec for 13 sec., then +4°F/sec for 13 sec.</p>	<p>The most extensive use of large diameter pipe for a high temperature (1000°F) sodium system was at Fermi. A 70 ft. run of 30 in. O.D. pipe was used between the reactor outlet and the IHX. Larger quantities of smaller pipe (16 in. O.D.) were used successfully in both Fermi and Hallam plants.</p> <p>State-of-the-Art reference:</p> <p>LMFBR Program Plan, Volume 3.</p>	<p><u>Engineering Data and Techniques</u></p> <p>The development of design criteria and analytical methods.</p> <p>The evaluation and/or development of piping support methods. Specific areas of concern: a) support of highly loaded large diameter, thin wall pipe, and b) support effect on piping system.</p> <p>The development of in-service inspection method and maintenance. Specific needs include: a) techniques, b) procedures, and c) inspection frequency.</p> <p>The development of fabrication technology in order to establish system with high degree of reliability. Areas to be covered: a) fabrication processes (shop and field), b) quality control, its limitations and system effects, and c) erection techniques and equipment requirements.</p>	<p>5.2.A</p> <p>5.2.B</p> <p>5.2.C</p> <p>5.2.D</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PRIMARY, INTERMEDIATE, AUXILIARY COOLANT SYSTEMS (Sections 6, 7, 10) Piping (General)(Paragraphs 6-2.1.5, 7-2.1.4, 10-2.1.5)			
<p>Piping, valves, and major components must have design life of thirty years; they must live through occasional leakage incidents.</p> <p>Definition of the conditions for rapid corrosion of steel by sodium with limited air access will assist in defining safety criteria to reduce the probability of such attack occurring on the reactor. The likely minimum time lapse between a small leak occurring and it becoming a catastrophic leak would be essential information to enable a leak detection apparatus specification to be drawn up.</p>	<p>There have been several instances of small sodium leaks in which massive and very rapid corrosion of external steel surfaces has occurred. The speed of this attack by sodium with limited air access is such that there is a danger of a small leak, if undetected for a short period, becoming catastrophically large. Such an attack occurred on a sodium expansion tank at HNDP, and there have been instances in the UK. A related phenomenon is the so-called "accelerated" corrosion of steel in sodium, intermediate in rate between that just described and "conventional" corrosion. This has been observed at sodium-cover gas interfaces.</p> <p>There is no quantitative information about catastrophic (very fast) corrosion.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Rate of attack of structural steels under defined conditions by sodium-sodium compound mixtures. Specific needs: a) Influence of temperature upon rate of attack, b) Influence of air components upon rate of attack--oxygen, moisture, possibly carbon dioxide, nitrogen, and c) effect of other materials on rate of attack--halides deriving from lagging materials, Al wrapping.</p> <p>Definition of conditions and sodium-sodium compound mixture producing maximum rate of attack.</p>	<p>3.7.A</p> <p>3.7.B</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
STEAM GENERATOR SYSTEM (Section 8) - Steam Generator (Paragraph 8-2)			
<p><u>General</u></p> <p>Type: Once-through, vertical, shell-and-tube design (single-wall tube)</p> <p>Thermal Rating: 1300 MWt</p> <p><u>Shellside</u></p> <p>Fluid: Non-radioactive sodium</p> <p>Flow Rate: 47.1×10^6 lb/hr</p> <p>Inlet Temperature: 967°F</p> <p>Outlet Temperature: 658°F</p> <p>Material: 304 SS</p> <p><u>Tubeside</u></p> <p>Fluid: Water/steam</p> <p>Flow Rate: 4.96×10^6 lb/hr</p> <p>Inlet Temperature: 500°F</p> <p>Outlet Temperature: 900°F</p> <p>Operating Pressure: 2770 psia</p> <p>Material: Incoloy 800</p>	<p>Current technology in operational, once-through, steam generators is represented by the steam generators employed in the Enrico Fermi Power Plant.</p> <p><u>General</u></p> <p>Type: Once-through, vertical shell-and-tube design (single-wall tube)</p> <p>Thermal Rating: 143 MWt</p> <p><u>Shellside</u></p> <p>Fluid: Non-radioactive sodium</p> <p>Flow Rate: 5.3×10^6 lb/hr</p> <p>Inlet Temperature: 820°F</p> <p>Outlet Temperature: 520°F</p> <p><u>Tubeside</u></p> <p>Fluid: Water/steam</p> <p>Flow Rate: 0.476×10^6 lb/hr</p> <p>Inlet Temperature: 380°F</p> <p>Outlet Temperature: 780°F</p> <p>Operating Pressure: 900 psig</p> <p>Sodium corrosion - very little data available on 304 SS and on Incoloy 800, but mostly obtained under conditions not representative of those expected in service.</p>	<p><u>Basic Data</u></p> <p>Sodium corrosion and erosion data on 304 SS and Incoloy 800 under representative service conditions, especially in areas of deposition.</p> <p>Data on mechanical properties in sodium of 304 SS and Incoloy 800 under representative service conditions. Main emphasis should be on creep, stress-rupture, stress relaxation and fatigue.</p> <p>Stress corrosion cracking resistance of Incoloy 800 in expected service and possible accident conditions.</p> <p>Fatigue resistance of Incoloy 800 in water/steam environments.</p> <p>Mechanical properties of tube-to-tube sheet weld in sodium.</p> <p>Sodium corrosion resistance of the tube-to-tube sheet weld.</p> <p>Effect of weld defects on properties of tube-to-tube sheet weld.</p> <p>Tube wastage from Na-water reaction.</p> <p>Effect of fabrication processes on above properties.</p> <p><u>Engineering Data and Techniques</u></p> <p>Design criteria based on a) the current state-of-the-art, and b) a review and assessment of the operational aspects of liquid-metal heat transfer components. This criteria must provide standards and guidelines for SG design (mechanical and functional), operation, manufacture, installation, maintenance, and component quality (i.e., inspection and quality assurance).</p> <p>Information, both analytical and experimental, to permit accurate thermal, hydraulic, and mechanical analysis. Information is principally needed for a) the verification, refinement and/or development of techniques to determine flow stability, flow distributions in steam generators, and b) the establishment of analytical techniques (based on experimental data) for predicting the consequences of a sodium/water reaction in a steam generator.</p>	<p>3.3.A</p> <p>3.3.B</p> <p>3.3.C</p> <p>3.3.D</p> <p>3.3.E</p> <p>3.3.F</p> <p>3.3.G</p> <p>3.3.H</p> <p>3.3.I</p> <p>5.6.C</p> <p>5.6.D</p> <p>5.6.F</p> <p>5.6.A</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
FUEL HANDLING (Section 11) - Fuel Handling Machines (Paragraph 11-2.3)			
<p>Vertical Fuel-Handling Machines - A high-precision bridge and trolley-type machine remotely operated, automatic and semi-automatically controlled from a shielded viewing area is required. The machine will position over a selected fuel element lower, grapple engage, raise and place the fuel element in a basket of the lateral fuel handling machine. It will also remove the fuel assembly from the basket and deposit it in the fuel storage tank.</p> <p>Lateral Fuel Handling Machine - a specially-designed machine operating above and in the fuel storage tank is required. The machine has a long boom at the end of which is a tiltable basket to receive fuel element. The boom enters a tube which connects the fuel storage tank and reactor vessel port, tilts the basket vertically to receive the element, returns the basket with the fuel to the horizontal, and withdraws the boom and basket to the spent fuel storage tank. In the spent fuel storage tank, the machines again rotate the basket vertical for removal of the fuel by the vertical machine.</p>	<p>Current fuel handling techniques use vertical handling of fuel with under sodium grapping, etc. No current technology uses a method where the fuel is rotated to the horizontal for withdrawal from the reactor. "Open pool" refueling, where high precision fuel handling machines work over a large open pool of sodium, is also unique for sodium cooled FBRs.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Data will be required on high precision equipment operating in a sodium vapor/argon environment. Data include sodium deposition at various temperatures, electrical voltage breakdown, lubricant effectiveness and life, bearing life, and seal effectiveness and life.</p> <p>Full scale machine tests are also required to develop high integrity, reliable fuel handling machines.</p>	<p>5.7.A 5.7.M</p> <p>5.7.M</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
FUEL HANDLING (Section 11) - Irradiated Fuel Assembly Gas and Liquid Cooling Systems (Paragraph 11-2.3)			
<p>Gas cooling of irradiated fuel assemblies will be required in the present 1000 MWe plant design when the assembly is lifted out of a sodium coolant for minimal inspection and transferred from bucket to can.</p> <p>A gas cooling system will help to eliminate the credible possibility of fuel melting during transfer under all possible operational malfunctions, including jamming of the fuel subassembly while partially submerged in sodium.</p> <p>Liquid cooling will be necessary in the buckets used to transfer spent fuel assemblies from the spent fuel pit to the inspection station. Liquid cooling systems will also be employed in the fuel transfer buckets and the spent fuel pits. These must be high integrity systems.</p>	<p>Both direct and indirect gas cooling has been used to permit handling of spent fuel, e.g., Dounreay subassemblies (2.5 KW decay heat load after shutdown) are cooled directly, while at Fermi assemblies are handled in gas cooled sodium pots.</p> <p>Experience has also been gained in gas cooling of fuel during transfer in gas cooled reactors. Little work other than theoretical studies has been done on gas cooling of fuel at the power densities of the Liquid Metal Fast Breeder Reactors.</p> <p>Liquid cooling techniques should be adequate to design a cooling system provided enough information on natural convection of liquid metals in buckets in pools is available.</p> <p>State-of-the-Art reference: LMFBR Program Plan, Volume 3.</p>	<p><u>Basic Data</u></p> <p>Radiation and convection heat transfer coefficients for stainless steel fuel pin arrays wetted with sodium and in a gas must be determined.</p> <p><u>Engineering Data and Techniques</u></p> <p>Natural convection heat transfer data in a liquid metal filled can containing a spent fuel assembly must be obtained for cases with and without cooling coils in the can.</p> <p>Heat generation rates in all subassemblies at the time of handling must be known as well as the heat capacity of the assembly.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Experimental heat transfer data are required to obtain necessary information to substantiate safety in fuel handling.</p>	<p>5.7.1</p> <p>5.7.1</p> <p>5.7.1</p> <p>5.7.1</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
FUEL HANDLING (Section 11) - Bearings (Paragraph 11-2.3)			
<p>Bearings for fuel handling machines which must support loads up to 1000 lbs. and operate reliably under sodium.</p>		<p><u>Experimental Verification and Operational Analysis</u></p> <p>The effect of sodium and inert gas purity on bearing behavior.</p> <p>The effect of oxygen on bearing behavior.</p> <p>The effects of friction factors and bearing stresses on specific wear rates.</p> <p>The effects of intermittent operation.</p> <p>Bearing life for extended operation.</p> <p>Bearing types and designs for refueling equipment.</p> <p>Design limitations - clearances, differential thermal expansion, etc. - consistent with materials compatibility.</p> <p>The effects of sodium vapor deposition on operation.</p> <p>Bearing life during operation through the sodium/inert gas interface.</p>	<p>5.7.B</p> <p>5.7.B</p> <p>5.7.B</p> <p>5.7.B</p> <p>5.7.B</p> <p>5.7.B</p> <p>5.7.B</p> <p>5.7.B</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
FUEL HANDLING (Section 11) - Mechanisms (Paragraph 11-2.3)			
<p>Mechanisms for hot cell re-fueling equipment that will operate reliably in sodium and inert gas/sodium environments. These include:</p> <ul style="list-style-type: none"> a) Vertical fuel handling machine grippers. b) Lateral or Bucket fuel handling machine basket releases. c) Gears. d) Racks. e) Linkages. 	<p>Many mechanisms required have already been developed in a mechanical sense, however, little has been done to determine their materials and operational compatibility with the working environment.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>The effects of sodium vapor deposition on operational performance.</p> <p>Performance parameters, e.g., thread design to minimize plugging by contaminants while still retaining acceptable load and backlash performance.</p> <p>Mechanism interactions must be known to permit a true specification of requirements, e.g., wear, for each item as it affects overall operation.</p> <p>Mechanism life for extended operation must be determined.</p> <p>The effects of intermittent operation must be determined.</p> <p>Design limitations - clearances, differential thermal expansion, etc. - consistent with required performance must be defined.</p> <p>Bearing life during operation through the sodium/inert gas interface.</p>	<p>5.7.A 5.7.C</p> <p>5.7.C</p> <p>5.7.C</p> <p>5.7.C</p> <p>5.7.C</p> <p>5.7.C</p> <p>5.7.B</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
FUEL HANDLING (Section 11) - Seals (Paragraph 11-2.3)			
<p>Inert gas hot cells require a high degree of seal reliability. Depending on cell pressures, one can get either leakage of contaminated gas out of the cell, thereby creating a severe hazard, or leakage of oxygen into the cell causing sodium contamination during refueling and increasing cell atmosphere purification costs. Seals include those required for viewing windows; electrical and gas penetrations; personnel, fuel and equipment hatches or ports, spent fuel and contamination storage pits; and personnel access suits and support systems.</p>	<p>Packing glands with and without an inert gas purge and metal bellows have been used in Fermi and EBR-II for sealing reciprocating motion. Problems encountered with the packing glands have been with excessive friction which has interfered with the push-pull capability of fuel handling equipment. The lifetime of metal seals has been limited by cyclic fatigue. They have also been subject to damage from sodium frosting (oxide clogs clearances with builds up from within). Static seals using organic or metallic gaskets, packing glands, and O-rings have not been as troublesome as dynamic seals.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Environmental (sodium vapor and inert gas) effects on seal materials and seal designs must be determined.</p> <p>Seal life during extended rotational operation must be determined.</p> <p>Seal life during intermittent operation must be determined.</p> <p>Design limitations - clearances, differential thermal expansion, etc. - consistent with required performance will be defined.</p>	<p>5.7.A 5.7.D 5.7.F 5.7.D 5.7.D 5.7.D</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
FUEL HANDLING (Section 11) - Electrical Components (Paragraph 11-2.3)			
<p>Electrical devices such as motors, switches, selsyns, connectors, and cables must be capable of reliable operation in a dry, hot inert atmosphere with a radiation background. Acknowledged problems involved with operating electrical equipment in inert atmospheres include: electric breakdown caused by lower dielectric strengths; overheating caused by poorer heat transfer properties; accelerated brush wear on commutating equipment; and sticking and pitting of contacts. Additional difficulties may be introduced by the presence of sodium vapor.</p>	<p>Electrical devices have operated successfully while isolated from the radioactive sodium vapor inert gas environment. As new fuel and components handling systems are developed a need to operate within this type of an environment will be required, however, reliable operation of current electrical devices in dry inert gases (argon or helium) is highly questionable. Cooling characteristics are such that devices in argon are limited to 65% of that in air.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>The effects of sodium on electrical devices must be assessed.</p> <p>Problems such as oxide buildup in and thermal distortion of electrical components must be defined.</p> <p>Voltage breakdown and brush wear characteristics in argon must be further defined.</p> <p>Life of operating components, e.g., motors, in a sodium vapor/hot, dry, argon atmosphere must be determined.</p>	<p>5.7.A</p> <p>5.7.E</p> <p>5.7.E</p> <p>5.7.E</p> <p>5.7.E</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
FUEL HANDLING (Section 11) - Lubricants (Paragraph 11-2.3)			
<p>Lubricants will be required for moving parts of fuel handling system components, hence must retain their lubricating qualities in a 150°F sodium vapor/dry argon atmosphere while under intense gamma radiation.</p>	<p>Although lubricated mechanisms have been successfully operated in inert gas environments, considerable uncertainties exist in lubricant additive properties in such conditions. Radiation and thermal effects on lubricants are well documented. Some data on volatility of lubricants (relevant to cover gas contamination) and on the solubility of argon in lubricants (relevant to foaming effects) are available. Almost nothing is known on the compatibility of lubricants with sodium.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Compatibility of lubricants and extreme pressure additives with sodium.</p> <p>Effectiveness of extreme pressure additives in inert atmospheres.</p> <p>Volatility of candidate lubricants in hot cell atmospheres.</p> <p>Solubility of argon in candidate lubricants and severity of foaming in typical applications.</p>	<p>5.7.F 5.7.G 5.7.G 5.7.G 5.7.G</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
FUEL HANDLING (Section 11) - Under-Sodium Viewing (Paragraph 11-2.3)			
<p>The critical item in the Under-Sodium Viewing System is the acoustical transducer. If this can be developed a system could be put together rather easily.</p> <p>a) The Under-Sodium Transducer must withstand the high radiation levels existing in the core. Present materials will probably age very quickly and suffer property changes. This ionizing radiation may provide paths for electrical breakdown of insulating materials, contributing noise in the readout devices.</p> <p>b) Strong thermal gradients will provide non-homogeneous indices of refraction and will result in undesired refractions of the acoustical beam.</p>	<p>Acoustical viewing is now used for underwater viewing and ocean bottom scanning. Object size to be viewed appears to be within the present state-of-the-art refinement. Acoustical lenses and scanning systems have been developed and should be adequate.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Object size and viewing area requirements must be determined.</p> <p>Scan rate must be determined.</p> <p>Transducer(s) configuration must be resolved.</p> <p>Radiation damage effects must be determined for transducer materials.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>The signal to noise ratio must be measured to aid in other design stages.</p>	<p>5.7.H</p> <p>5.7.H</p> <p>5.7.H</p> <p>5.7.F 5.7.H</p> <p>5.7.H</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
FUEL HANDLING - Viewing System (Paragraph 11-2.3)			
<p>a) Telescopes</p> <p>b) <u>Binooculars</u></p> <p>Telescopes, binoculars, etc., shall be of the offset type to prevent radiation streaming. Lenses shall be the cerium - stabilized or equal type of non-browning lenses. Magnification and resolution should be as required for the particular application cited.</p> <p>c) <u>Shielded Windows</u></p> <p>Shielded windows shall attenuate gamma activity to acceptable level without attenuating visible lighting by too large a factor. Window transmission characteristics should match the high intensity lighting and the television systems' spectral characteristics. Material used shall not suffer radiation damage at levels anticipated.</p> <p>d) <u>Closed Circuit Television</u></p> <p>Closed circuit television systems are presently available for use in cells with similar radiation and lighting problems. External cooling would be required on all known equipment.</p>	<p>Telescopes, binoculars, etc. are presently available from several sources. Special ones could be constructed for specific applications without difficulty.</p> <p>Presently available shielded windows are generally adequate, and more desirable ones are being proposed on a development basis.</p> <p>The closed circuit television system must be capable of operating in a 150°F ambient temperature and, an inert, dry argon atmosphere. The lenses, and circuitry, must be able to withstand the high radiation level, while the circuitry must be kept to a minimum and be modular or otherwise suited to remote maintenance.</p>	<p><u>Engineering Data and Techniques</u></p> <p>The required resolution must be determined.</p> <p>All components must be designed for long life cycles in the 150°F dry argon atmosphere.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Radiation effects on component life must be determined.</p> <p>The effect of sodium vapor deposition on resolution and component life must be determined.</p> <p><u>Engineering Data and Techniques</u></p> <p>The effect of hot cell radiation levels on window life.</p> <p>The effect of window transmission characteristics on television system spectral characteristics.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>The effect different types of hot cell windows have on viewing clarity when the reference lighting concept is utilized.</p> <p>The effect of sodium vapor deposition on viewing and on window life.</p> <p><u>Engineering Data and Techniques</u></p> <p>The required resolution must be determined.</p> <p>Television systems must be designed for long life cycles in the 150°F dry argon atmosphere.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>The effect of sodium vapor deposition on resolution and lens life must be determined.</p> <p>The light intensity and spectral characteristics must be determined.</p>	<p>5.7.K</p> <p>5.7.F</p> <p>5.7.K</p> <p>5.7.K</p> <p>5.7.A</p> <p>5.7.K</p> <p>5.7.F 5.7.K</p> <p>5.7.K</p> <p>5.7.K</p> <p>5.7.A</p> <p>5.7.K</p> <p>5.7.K</p> <p>5.7.K</p> <p>5.7.A</p> <p>5.7.K</p> <p>5.7.K</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
FUEL HANDLING (Section 11) - Manipulator System (Paragraph 11-2.3)			
<p>The manipulators must cover all work areas in the hot cell from an elevation of 36 ft. above the floor to 3 ft. below the floor.</p> <p>A standard manipulator arm with shoulder, elbow, wrist, and hand motion is required.</p> <p>The manipulators will be controlled from a remote console, positioning will be by visual access.</p> <p>The manipulator must operate reliably in a dry argon/sodium vapor environment and perform all non-contact maintenance functions.</p> <p>All manipulator parts must be remotely removable and replaceable.</p>	<p>Significant design and development programs for manipulators have been carried out for a number of years at ANL. As a result a number of manipulator types such as unilateral remote, mechanical master-slave, and electro-mechanical master-slave are now operational.</p> <p>Manipulators of all the above types are manufactured by companies such as PAR and CRL. In many cases these commercially available manipulators are based upon ANL designs.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Information is required on how the dry argon/sodium vapor environment will affect the lubrication, cooling, and electrical properties of the manipulator components.</p> <p>Much work is required in increasing load and motion capability of the multi-purpose manipulators and obtaining better resolution on the television system used for viewing in-cell operations.</p> <p>The extent of remote maintenance operations must be known to guide manipulator development.</p>	<p>5.7.F 5.7.L</p> <p>5.7.L</p> <p>5.7.L</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PLANT SERVICE SYSTEM (Section 12) - Manned Access System (Subsection 12.7)			
<p>The 1000 MWe Plant will require both remote and contact maintenance. Manned entry into the inerted hot cell environments shall be done by personnel dressed in an enclosed suit serviced by an air supply and return system. The entire in-cell Manned Access System must be leak-tight to maintain the stringent atmosphere controls on oxygen.</p>	<p>Very little work has been done on Manned Access Systems especially for reactor plants. SEFOR utilizes a Manned Access System but little data is available. UNC, ORNL, and the Universal Cyclops Specialty Steel Corp. have had experience with sophisticated access systems but none of those have cells with a high purity inert atmosphere, radioactive fission products and sodium vapor.</p>	<p><u>Engineering Data and Techniques</u></p> <p>The types of contact maintenance that must be performed and the location of components that will probably be serviced.</p> <p>The phases of plant operation that will require manned access, i.e., shutdown with plug off, during operation, shutdown with plug on, etc.</p> <p>Acceptable locations for access doors in the containment structure and restrictions on access lock construction, if any.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>The effect of fission products, sodium vapor and the dry hot inert gas atmosphere on system components, e.g., suit, helmet, hose, etc.</p>	<p>5.7.J</p> <p>5.7.J</p> <p>5.7.J</p> <p>5.7.F 5.7.J</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PLANT SERVICE SYSTEMS (Section 12) - Decontamination and Remove Maintenance Systems: Wash Down Equipment (Subsection 12.9.2)			
<p>Decontamination of components removed from the reactor from adsorbed fission products and active corrosion products and, in case of fuel failure, fuel. Reduction to a level where arduous handling procedures are not required. Decontamination criteria not specified.</p>	<p>A variety of processes have been used, but their effectiveness with respect to fission products and active corrosion products is not sufficiently established. After a fuel cladding failure in BR-5, absorbed fission products were little effected by steam cleaning and water flushing. A more drastic treatment with acid and $KMnO_4$ solutions was required to reduce activity to an acceptable figure.</p> <p>No state-of-the-art descriptions are given in LMFBRR Program Plan Volume 5 (Sodium Technology).</p>	<p>Experimental Verification and Operational Analysis</p> <p>Efficiency of candidate sodium decontamination in removing absorbed fission products and active corrosion products from component surfaces.</p> <p>Establish an acceptable method of removing such absorbed activity.</p> <p>Determine extent of penetration of activity into material structure.</p> <p>Any possible plant system modifications necessary for implementation of decontamination procedures.</p>	<p>7.3.E</p> <p>7.3.E</p> <p>7.3.E</p> <p>7.3.E</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PLANT SERVICE SYSTEMS (Section 12) - Sodium Purification System (Paragraphs 12.1-2, 12.5-2)			
<p>a) <u>Impurities</u></p> <p>Purification of initial sodium charge by filtering and cold trapping to produce required sodium plugging temperature <325°F.</p> <p>Capability for maintaining sodium coolant purity requirements through use of cold traps and hot traps with capacity for one day system volume turnover rate.</p> <p>Establish and maintain sodium purity at:</p> <p>Oxygen ≤10 ppm by wt. Carbon ≤20 ppm by wt. Nitrogen ≤20 ppm by wt.</p> <p>Hydrogen; saturation temperature <325°F.</p> <p>Plug forming impurities; saturation temperature 325°F.</p> <p>Parallel R & D programs call for development of fuel cladding. If a refractory metal alloy is used, then the purity specification is not good enough. Probably oxygen would have to be: <5ppm by wt.</p>	<p>Current technology is limited to the use of empirically designed cold trap systems for both small and large engineering systems and to the applications of zirconium and stainless steel filled hot traps used in small engineering and experimental systems.</p> <p>Lower purity levels (if required by cladding developments) require reaction trapping either by use of soluble getters (e.g., Ca) or by hot trapping (usually performed by Zr for oxygen, stainless steel for carbon). There exists considerably more experience of hot trapping than of soluble gettering. Experience of oxygen and carbon trapping in large scale operations is limited and mainly derives from SRE.</p> <p>State-of-the-Art reference: LMFBR Program Plan, Volume 5.</p>	<p><u>Basic Data</u></p> <p>Solubility of impurities in liquid sodium.</p> <p>Quantitative information on precipitation kinetics involved in oxide removal in cold trap systems, and carbon removal in hot trap systems.</p> <p>Properties of precipitates formed in cold and hot traps under various conditions of flow, temperature and time.</p> <p>Effect of minor impurities including fission products on the precipitation of oxide from liquid sodium.</p> <p>Natural circulation heat transfer data for liquid sodium.</p> <p>Basic information on materials which are potential soluble getters if required as a result of cladding development</p> <p>Distillation data for sodium including the effect of rate, pressure, temperature and materials of construction on composition of distillate and residue.</p> <p>Basic information on impurity levels anticipated and ability to measure them</p> <p><u>Engineering Data and Techniques</u></p> <p>The effect of residence time, geometry, nucleation sites, and temperature control on cold and hot trap operation.</p> <p>Computational methods and codes for evaluation of cold and hot trap performance</p> <p>Systematic design criteria for hot traps including optimum reaction temperature, relative rate functions for the various refractory materials including saturation phenomena effects of hot trapped sodium on structural materials and assessment of particulate addition to the sodium.</p> <p>Quantitative data for the design of soluble getter hot trap systems including dilute getter concentration measurement and maintenance in sodium, getter oxide removal, transport of container materials under thermal gradients in soluble getter system, and hydrogen concentration effects, limitations and control.</p> <p>Effectiveness of purification by distillation determined by modern analytical methods.</p> <p>Experimental data for evaluation of conceptual designs for filtration, settling and centrifuges to remove particulate matter from sodium.</p>	<p>7.1.A, 7.1.B, 7.5.H</p> <p>7.1.A 7.1.B</p> <p>7.1.A 7.1.B</p> <p>7.1.A 7.1.B</p> <p>7.1.A</p> <p>7.1.B</p> <p>7.1.D</p> <p>7.5.B, 7.5.C, 7.5.D, 7.5.E, 7.5.F</p> <p>7.1.A 7.1.B</p> <p>7.1.A 7.1.B</p> <p>7.1.B</p> <p>7.1.B</p> <p>7.1.D</p> <p>7.1.C</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PLANT SERVICE SYSTEMS (Section 12) - Sodium Systems(Paragraphs 12.1-2, 12.5-2) (Continued)			
<p><u>b) Fission products and active corrosion products</u></p> <p>Continuous removal is required to comply with maintenance and safety requirements. Levels not specified (but the lower the better).</p>	<p>No state-of-the-art descriptions are given in LMFBF Program Plan, Volume 5 (Sodium Technology)</p> <p>Much of the available data on fission product behaviors has arisen from a study of accident conditions, simulated and real e.g. from the fuel element cladding failure in SRE. The data consequently is concerned with a sudden release of gross quantities of fission products instead of lower-level steady-state conditions such as might exist with reprocessed fuel. It has been pointed out that existing data from reactors under normal running conditions is qualitative in nature.</p> <p>Some of the basic work of LASL and BNL has been published.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Pilot plant operation to demonstrate successful performance of small scale conceptual designs and optimized final design for:</p> <ol style="list-style-type: none"> Cold traps Hot traps Soluble gettering system Mechanical separation of particulate matter from sodium. <p><u>Basic Data</u></p> <ol style="list-style-type: none"> Nature and solubility of fission products in sodium Interaction of fission products with impurities in sodium and containment materials Identification of materials having high adsorption for fission products and compatible with sodium, as candidate materials for fission product control <p><u>Engineering Data and Techniques</u></p> <ol style="list-style-type: none"> Expected rate of release of fission products and activated corrosion products to primary coolant Acceptable level of radioactivity in primary sodium and on system surfaces Effectiveness of various methods for removal of fission products and active corrosion products from sodium, i.e., cold traps, hot traps. Sodium-cover gas interface fission product release studies; distribution of fission products between sodium phase and cover gas <p><u>Experimental Verification and Operational Analysis</u></p> <ol style="list-style-type: none"> Proof tests of selected concept for fission product management in sodium 	<p>7.1.A 7.1.C 7.1.B 7.5.G</p> <p>7.3.B 7.5.F 7.3.B 7.3.B</p> <p>7.3.A 7.3.A 7.5.I 7.3.B 7.5.G 7.3.C 7.3.B 7.3.D</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PLANT SERVICE SYSTEMS (Section 12) - Gas Purification (Paragraph 12.2-2)			
<p>Removal of trace amounts of oxygen and moisture from inert (argon) gas.</p> <p>Removal of gaseous fission products from the primary cover gas (argon) by activated carbon adsorption beds at cover gas circulation rate in the range of 100 to 300 lb/hr, and a 10 day fission gas capacity.</p> <p>Removal of sodium vapor and aerosol from primary and secondary cover gas.</p> <p>Removal of non-radioactive impurities from fuel handling and maintenance area inert gas.</p>	<p>There are several practical procedures for separation of fission gases, but adsorption on activated carbon has been used primarily in this field. This concept has been successfully used in gas cooled reactors (Peach Bottom, Dragon) but with helium gas, not argon, and conditions were considerably different.</p> <p>Sodium vapor traps have found almost universal use in liquid metal reactor systems where large volumes of gas are transferred. However, operation has been inconsistent and, in general, equipment oversized to compensate for lack of design criteria. Development has not proceeded from preliminary work.</p> <p>General inert gas blanket technology has been neglected in reactor systems. Most information available is from glove box and inert cell technology and is not necessarily applicable.</p> <p>State-of-the-Art reference:</p> <p>LMFBR Program Plan, Volume 5.</p>	<p><u>Basic Data</u></p> <p>Adsorbent characteristics including static equilibrium data for krypton and xenon, heat conductance and high temperature stability. 7.2.A</p> <p>Significant factors which contribute to the formation of sodium aerosols at liquid-gas interfaces. This would include the effect of liquid temperature, velocity of gas over the surface, temperature gradient in the gas and similar factors. 7.2.C</p> <p>The effect of nitrogen impurity in the cover gas including solubility in liquid sodium, and rate of nitriding versus liquid sodium temperature. 7.2.B</p> <p>The effect of trace impurities in the cover gas such as carbon monoxide, carbon dioxide and hydrocarbons, and how these impurities may be removed. 7.2.B</p> <p>Factors controlling the condensation of sodium vapor at low partial pressure in cover gas. 7.2.C</p> <p>Adsorbent characteristics for the removal of sodium vapor. 7.2.C</p> <p><u>Engineering Data and Techniques</u></p> <p>Dynamic equilibrium coefficient for argon, krypton, xenon and other trace gases on adsorbents in relation to partial pressure, flow velocity and temperature. 7.2.A</p> <p>Heat transfer and temperature gradients in adsorbent beds caused by decay heat within the bed and cooling external to the bed. 7.2.A</p> <p>Adsorbent bed regeneration procedures including required conditions of temperature, pressure and flow rate. 7.2.A</p> <p>Effect of solid decay products such as cesium on adsorbent bed capacity. 7.2.A</p> <p>Economic optimization of adsorbent bed temperature versus bed size. 7.2.A</p> <p>Degradation of adsorbent bed due to many cycles of adsorption and reactivation. 7.2.A</p> <p>The effects of transients in flow, temperature, and composition on bed performance. 7.2.A</p> <p>Evaluation of NaK scrubber, catalytic reactor, chemical reaction and cryogenic distillation concepts for the removal of trace quantities of impurities in cover gases. 7.2.B</p> <p>Means for suppression of aerosol formation in liquid sodium systems. 7.2.C</p>	

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PLANT SERVICE SYSTEMS (Section 12) - Gas Purification (Paragraph 12.2-2)			
		<p><u>Experimental Verification and Operational Analysis</u></p> <p>Pilot plant operation to demonstrate successful performance of small scale conceptual designs and optimized final design for:</p> <ul style="list-style-type: none"> a) Fission product gas adsorption system. b) Sodium-aerosol vapor trap. c) Gas purification system. 	<p>7.2.A</p> <p>7.2.B</p> <p>7.2.C</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PLANT SERVICE SYSTEMS (Section 12) - Process Instrumentation and Control: Sodium Purification System (Paragraph 12.1-2.4)			
<p>Establish and maintain sodium purity at:</p> <p>Oxygen <10 ppm by wt.</p> <p>Carbon ≤20 ppm by wt.</p> <p>Nitrogen ≤20 ppm by wt.</p> <p>Hydrogen; saturation temperature <325°F.</p> <p>Plug forming impurities; saturation temperature <325°F.</p> <p>For adequate sodium quality control, sampling and analysis of the active sodium is required. In-line instruments giving continuous readout are required.</p> <p>Quick detection of steam generator to secondary sodium system leaks is mandatory.</p>	<p>For sodium analysis, amalgamation methods are now routine for oxygen and two apparent forms of hydrogen. Various other procedures based on diffusion for hydrogen and combustion for carbon are used. When these methods are considered for active sodium, the handling difficulties are seen to be formidable, and alternative methods must be sought. Another method under current development is activation analysis, but again this would be extremely difficult to apply to active sodium because of the difficulties of activity discrimination; the method also suffers from lack of sensitivity. Spark source mass spectrometry with remote sample handling is being developed by W. It is not yet certain however, that such a technique would be suitable for non-metallic impurities; also the extent of spectrometer contamination needs assessment. Vacuum distillation is a further method with potentialities. In the UK, it has been used for elemental carbon, total carbon, carbonate, and metallic impurities.</p> <p>The additional requirement for active sodium is that the distillation should be performed in the actual sampler so that most activity is left behind when the residue is removed. LASL appears to favor vacuum distillation and has developed an in-line unit which takes single samples.</p> <p>In-line meters for oxygen, carbon, and hydrogen are in various stages of development.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Develop adequate sampling and analysis for active sodium. Specific needs include:</p> <ol style="list-style-type: none"> A sampler/distillation unit capable of taking representative samples, transfer and distillation of sodium as a remote operation, disposal of distillate back to circulating system, or waste as a remote operation. Parallel development of spark source mass spectrometry with remote sample handling, ability to measure non-metallics, assessment of active contamination problem. <p>Develop in-line meters for oxygen, carbon and hydrogen.</p> <p><u>Oxygen Meter</u> - Specific needs include:</p> <ol style="list-style-type: none"> Improvements in ceramic electrolyte in the areas of purity, preparation methods. Electrolyte protection methods/devices. Influence of other sodium impurities on meter behavior. General assessment of performance of an improved meter with respect to sodium temperature, ambient temperature, temperature coefficient speed of response, relationship between meter output and conventional analysis. <p><u>Carbon Meter</u> - When UNC meter is commercially available, an assessment of performance in areas determined by the course of current development.</p> <p><u>Hydrogen Meter</u> - When APDA meter is commercially available, an assessment of performance in areas determined by the course of current development.</p> <p>Develop adequate back-up impurity instrumentation. The UK continuous indication plugging meter is thought to be obtainable commercially. Performance assessment will be necessary including:</p> <ol style="list-style-type: none"> Ability to measure very low saturation temperatures (approaching the freezing point of sodium). Correlation between saturation temperature and conventional analysis during clean-up procedures. 	<p>7.5.A</p> <p>7.5.F</p> <p>7.4.A, 7.5.B</p> <p>7.4.B 7.5.C</p> <p>7.4.C, 7.5.D</p> <p>7.4.D</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PLANT SERVICE SYSTEMS (Section 12) - Process Instrumentation and Control: Sodium Purification System (Paragraph 12.1-2.4) (Continued)			
	<p>The oxygen meter based on a galvanic cell with solid oxide electrolyte has been established the longest and is marketed by UNC. Much user experience has accumulated. The overall impression is that the oxygen meter still requires much development. Presence of considerable impurities in the electrolyte has recently been demonstrated, and these undoubtedly detract from meter performance. W is pursuing an independent development of the oxygen meter based on a commercially successful W gaseous oxygen cell.</p> <p>APDA have recently reported on a hydrogen diffusion meter which appears very promising. After diffusing from the sodium through a Ni/Pd membrane, hydrogen is catalytically induced to react with flowing oxygen by the Pd. The resultant moisture content of the gas stream is measured. UNC have also started work on an electrochemical hydrogen meter. The difficulties of working with a lithium hydride electrolyte with its high dissociation pressure appear great.</p> <p>UNC are also developing the carbon meter. It relies on diffusion of carbon from sodium through an iron membrane into a decarburizing atmosphere of hydrogen in moist argon. This converts the carbon to gaseous form, principally methane, which is determined by a flame ionization detector. Several probes are currently being tested in sodium.</p>	<p>c. Ability to measure saturation temperatures of multiple impurities when present; ability to work in conventional plugging meter fashion (cyclic).</p> <p>Identification of the materials precipitated from sodium by cooling below the appropriate saturation temperature.</p>	7.5.G

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
PLANT SERVICES SYSTEMS (Section 14) - Process Instrumentation and Control: Sodium Purification System (Paragraph 12.1-2.4) (Continued)			
	<p>Two unspecific guides to impurity level at present in use are the resistivity meter and plugging meter. Continuous-indication plugging meters have been developed in France and the UK. Plugging meters have received attention at LASL who have shown there is a super-saturation requirement for impurity nucleation and that plugging temperature is not the best measure of saturation temperature. The continuous-indication meter works in a partially plugged state; no nucleation is called for, hence saturation temperature is indicated continuously.</p> <p>Plugging meter precipitation of second and third impurities has now been observed on numerous systems including reactors. Even though it is not known what the impurities are, it should be noted that the plugging meter "resolves" them whereas the resistivity meter does not. The effect of the impurities on corrosion of reactor materials is not certain, but GE have expressed a tentative opinion that at the levels observed they are not significant. Plugging of reactor components may be of greater significance since saturation temperatures greatly in excess of cold trap temperature have been observed. A continuous-indication meter operates in such a way as to lock on to the highest saturation temperature. Intermittant measurement remains useful, therefore, under circumstances where several materials having different saturation temperatures are present.</p>		

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Nuclear (Paragraph 14-2.4)			
<p>a) <u>Low Power Neutron Detector</u> Type: Pulse Chamber (U-238 coated) located in lower axial blanket and retracted at higher power levels. Range: Source to 100 KWth Environment: ~ 100 nv at 1 mw, 5×10^{13} - 5×10^{14} nv at full power. Immersion in liquid sodium at 7-800°F (1100°F accident condition) and γ dose rate 10^7 R/hr. Low sensitivity, fast response, long lifetime (>1 year).</p> <p>b) <u>Intermediate Lower Neutron Detector</u> Type: Compensated ion chamber located in neutron shield. Range: 10KW - 1000MW Environment: $\sim 3 \times 10^{12}$ nv thermal neutrons at full power. Conditions: 550-600°F and γ dose rate 10^5 - 10^6 R/hr. Fast response and long life (>1 year).</p> <p>c) <u>High Power Neutron Detector</u> Type: Ion chambers. Range: 1 MWth - 1000 MWth Site Flux: $\sim 10^{11}$ nv thermal neutrons at full power. Conditions: 350-400°F and γ dose rate 10^5 - 10^6 R/hr. Fast response and long life (>1 year).</p>	<p>No instruments are available for operation under these conditions. Instruments are currently available for operation at 600°F. Development and testing is planned for devices operating at 1100-1200°F.</p> <p>EBR II and Fermi use out-of-vessel instruments over this range.</p> <p>EBR II and Fermi use out-of-vessel instruments over this range.</p>	<p><u>Engineering Data and Techniques</u> Specification of flux levels over instrument site region. Data concerning materials and parts for instruments, including chamber body, neutron sensing, gas fill, γ tolerance, neutron spectra. Cables - See Signal Transmission Components.</p> <p><u>Experimental Verification and Operational Analysis</u> Lifetime and sensitivity data from testing prototypes.</p> <p><u>Engineering Data and Techniques</u> Specification of flux levels over instrument site region.</p> <p><u>Experimental Verification and Operational Analysis</u> Lifetime and sensitivity data from testing of wide range instruments operating at these γ fluxes for cooled and uncooled concepts. Extension of data for in-vessel low power neutron detector to intermediate power ranges for the event that a modular core is not amenable to out-of-vessel monitoring.</p> <p><u>Engineering Data and Techniques</u> Specification of flux levels over instrument site region.</p> <p><u>Experimental Verification and Operational Analysis</u> Lifetime and sensitivity data from testing of instruments operating at these γ fluxes for cooled and uncooled concepts. Extension of data for in-vessel intermediate power neutron detector to high power ranges in the event that a modular core is not amenable to out-of-vessel monitoring.</p>	<p>6.1.A 6.1.A 3.2.C 6.1.A 6.1.A</p> <p>6.1.A</p> <p>6.1.A</p> <p>6.1.A</p> <p>6.1.A</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Nuclear (Paragraph 14-2.4) (Continued)			
<p>d) <u>Primary Sodium Gamma Monitoring</u> Type: Ion chamber located in cover gas.</p> <p>Conditions: 1200°F, saturated with sodium vapor, γ dose rate $\sim 10^6$ R/hr.</p> <p>e) <u>Signal Conditioning Equipment</u> Counter operating over a wide range, up to at least 10^6 counts per second.</p> <p>Reliable in an environment near the reactor (in order to minimize cable length).</p>	<p>In-vessel instruments are protected by and cooled in thimbles.</p> <p>Counters currently rated at about 10^5 counts per second in practical applications. This disagrees with the LMFBR Program Plan, Volume 4.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Specification of normal and fault γ flux levels in cover gas.</p> <p>Data concerning materials and parts for instruments including chamber body, γ sensing, gas fill.</p> <p>Instrument design and logic development for high rate and wide range counting techniques against high γ backgrounds.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Lifetime and sensitivity data from testing prototype.</p> <p>Sensitivity and reliability data from testing prototypes.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Design and counter capable of 10^6 cps in high gamma background.</p> <p>Performance and lifetime data for 10^6 cps counter.</p>	<p>6.1.A</p> <p>6.1.A</p> <p>6.1.A</p> <p>6.1.A</p> <p>6.1.A</p> <p>6.1.A</p> <p>6.1.A</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Temperature Sensing Devices (Paragraphs 14-2.4)			
<p>Type: Thermocouples (Chromel-Alumel).</p> <p>Range: Up to 1200°F.</p> <p>Conditions: Immersion in liquid sodium with a γ dose rate of 10^6 R/hr.</p> <p>Fast response time with little γ heating, long life and resistance to thermal shock.</p>	<p>EBR II used chromel-alumel thermocouples and platinum resistance thermometers. The latter were used to calibrate the thermocouples.</p> <p>Fermi used iron-constantan thermocouples at lower temperatures.</p> <p>Tests have been made up to 2000°F and a life of 10,000 hrs. at 2200°F.</p> <p>Thermocouples for use up to 4500°R have been developed (tungsten-rhenium) for short life applications in NERVA Program.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Performance and lifetime data from statistically significant samples including stability under thermal cycling conditions in an LMFBR environment.</p> <p>Methods and techniques for positioning and servicing thermocouples at core exit locations.</p> <p>Testing of thermocouples in a fuel assembly exit mock-up to show effect of position on temperature indicated.</p> <p>Performance and reliability data for cold junction reference box.</p> <p>Performance and lifetime data for alternative types of device, e.g., resistance thermometers and ultra-sonic methods.</p>	<p>6.1.B 3.2.C</p> <p>6.1.B</p> <p>6.1.B</p> <p>6.1.B</p> <p>6.1.B</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Coolant Flow (Paragraph 14-2.4)			
<p>Type: Venturi and measurement of differential pressure in a bypass line. The differential pressure is measured by relating the flow to the power required by an E.M. pump to produce a null flow condition which is detected by an E.M. flowmeter.</p> <p>Range: 0 to 120,000 gpm in a pipe of diameter about 4 feet.</p> <p>Conditions: 1200°F and γ dose rate of 10^6 R/hr.</p>	<p><u>EBR II</u></p> <p>Permanent magnet E.M. flowmeter (~ 14 in. diameter).</p> <p><u>PFR</u></p> <p>Saddle coil E.M. flowmeter.</p> <p>Transit time flowmeter (~ 14 in. diameter).</p> <p><u>Fermi</u></p> <p>Permanent magnet E.M. flowmeters.</p> <p><u>Hallam</u></p> <p>Permanent magnet E.M. flowmeters (~ 14 in. diameter).</p> <p>Noise effects require damping out and accuracy may be as poor as $\pm 10\%$.</p> <p>Other methods include:</p> <ul style="list-style-type: none"> Turbines (small size) Transit time (PFR) Eddy currents Ultrasonics 	<p><u>Engineering Data and Techniques</u></p> <p>Investigation of requirements for ultra-sonic flow instrument and its calibration. These include materials for ultra-sonic transducers, cooling and attachment requirements, electronic circuitry and study of the flow characteristics of the instrument.</p> <p>Studies for alternative instruments, e.g., thermal and mechanical tests on magnetic materials, bearing materials, high temperature pick-up coils.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Calibration and sensitivity tests of a large diameter Venturi device (about 4 ft.) including the effects of pipe circuit configurations and bypass circuits (sodium and electronic).</p>	<p>6.1.C 3.2.C</p> <p>6.1.C</p> <p>6.1.C</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Strain Gauges (Paragraph 14-2.4)			
<p>Condition: Operation at temperatures up to 1200°F in air or under sodium. γ dose rates up to 10^6 R/hr. Capability of withstanding thermal shock.</p>	<p>Gauges capable of operation in air at 900°F are available. However, gauges have been developed for the NERVA Program for higher temperatures, but lifetime is only moderate (about one month).</p> <p>Liquid Metal Engineering Center reports a gauge suitable for 1200°F operation.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Continuation of experimental data acquisition for promising gauge designs including developments of gauges such as those used in the NERVA Program.</p> <p>A feasibility study of new methods of gauge attachment and new measurement techniques.</p>	<p>6.1.E</p> <p>6.1.E</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Level Detectors (Paragraph 14-2.4)			
<p>Type: Inductance probe and bridge.</p> <p>Range: Up to about 3 ft.</p> <p>Conditions: Immersion in liquid sodium at temperatures up to 1200°F and γ dose rates of 10^7 R/hr. for in-vessel applications.</p>	<p><u>EBR II</u></p> <p>Differential pressure measurement using a NaK filled transducer.</p> <p><u>Fermi</u></p> <p>Pneumatic bellows detectors located in the cover gas and under the sodium.</p> <p><u>SNAP</u></p> <p>ORNL "J" tubes calibrated against a movable inductance probe.</p> <p>Sodium Components Test Installation (SCTI) - four gauges based on radiation, displacement float, bubbler and induction coil.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Data concerning suitable materials for and designs of coils.</p> <p>Data concerning wetting phenomena in relation to inductance and other types of probe.</p> <p>Lifetime and sensitivity data from statistically significant samples.</p> <p>Fundamental data for other principles capable of use as back-up instrumentation. These may be based on differential pressure measurements, floats, resistance probes, ultra-sonics, radiation or micro-wave techniques.</p>	<p>6.1.F 3.2.C</p> <p>6.1.F</p> <p>6.1.F</p> <p>6.1.F</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Leakage Detection (Paragraph 14-2.4)			
<p><u>a) Sodium</u></p> <p>To detect changes in fission product concentration in the sodium from those levels associated with normal leakage from vented fuel.</p>	<p><u>EBR II</u></p> <p>Routine spectrochemical and radio-chemical analyses are performed. There is a long time lag before failures are detected.</p> <p><u>PFR</u></p> <p>Continuous sodium sample monitored for delayed neutrons. Individual subassembly monitoring if a failure is detected.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Study of fission product levels and distribution in primary circuit sodium resulting from various types of fuel failure for closed and vented fuels.</p> <p>Study of present techniques leading to refinements of the most promising. This will include faster methods of sampling to reduce the response time of the system and improve the signal/background noise ratio for delayed neutron and γ spectroscopic methods.</p> <p>The response time of this systems is necessarily longer than that of the above methods. A study is required to determine whether it is a suitable back-up system compared to a separate liquid sodium sampling system. Its sensitivity in a vented fuel reactor may be poor.</p>	<p>6.1.G</p> <p>6.1.G</p> <p>6.1.G</p>
<p><u>b) Cover Gas</u></p> <p>To detect changes in fission product concentration in the cover gas from those levels associated with normal leakage from vented fuel.</p>	<p><u>EBR II</u></p> <p>Detection of Xe and Kr by gas chromatograph proposed.</p> <p>Response rate is slow.</p> <p>Gas cooled reactors (UK)--all channels have samples withdrawn for detection of fission products.</p>	<p><u>Experimental Verification and Operational Analysis</u></p> <p>Sensitivity and reliability data for electronic circuitry is required.</p>	<p>6.1.G</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Signal Transmission Components (Paragraph 14-2.4) (Continued)			
<p>a) <u>Electrical Connectors</u> Methods and equipment are required to connect electrical transmission lines in such a manner that signal attenuation is minimal and there is no cross-talk, leakage, thermal E.M.F.'s and noise to reduce sensitivity.</p> <p>b) <u>Electrical Cables</u> Types required: Low leakage coaxial cable for nuclear instrumentation, multiconductor cable for electrical devices (e.g., motors and relays), and low-impedance signal devices (e.g., thermocouples and strain gauges).</p> <p>Capabilities: Immersion in liquid sodium at temperatures up to 1200°F and γ dose rates of 10^6 R/hr. These should be as flexible as possible (e.g., for winding on a drum), capable of transmitting a signal with minimal attenuation over distances of at least 30-40 feet and have a long life.</p> <p>c) <u>NaK Transmission Capillaries and Sample Lines</u> These are required to operate throughout reactor life without blockage or leakage.</p>	<p>There is no capability at present for under sodium connectors. Those designated for use in cover gas blankets are bulky and complex. Techniques are available for low temperature, high pressure salt water and for gases at temperatures up to about 400°F.</p> <p>Cables capable of withstanding the high temperature are claimed to exist by the manufacturers, but none are rated for the sodium conditions.</p> <p>State-of-the-Art references:</p> <p>LMFBR Program Plan, Volume 4.</p> <p>ANL-7380: Proc. Symp. LM Instrumentation and Control.</p> <p>In several liquid metal plants and rigs (e.g., EBR II), the capillaries have become blocked.</p> <p>State-of-the-Art reference:</p> <p>ANL-7380: Proc. Symp. LM Instrumentation and Control.</p>	<p><u>Engineering Data and Techniques</u></p> <p>For connectors under sodium or gases, a feasibility study is required to evaluate methods and economic incentive for a development program.</p> <p>Due to the fragile nature of available ceramic connectors, the methods studied will include the cooling of organic insulation connectors to below 400°F and preferably to less than 250°F.</p> <p><u>Engineering Data and Techniques</u></p> <p>Materials studies are required for compatible cladding, insulation, sealants and conductors.</p> <p>Methods for multiplexing under in-vessel conditions should be studied in order to reduce in-vessel cabling.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Cable designs utilizing the materials indicated by the above study will be manufactured and tested in statistically significant quantities.</p> <p><u>Engineering Data and Techniques</u></p> <p>Study of causes of failure. Development of methods for prevention of blockage, line clearance and replacement.</p>	<p>6.1.I</p> <p>6.1.I</p> <p>6.1.I</p> <p>3.2.C</p> <p>6.1.I</p> <p>6.1.I</p> <p>6.1.I</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Reactor Control System (Paragraph 14-2.1)			
<p>To provide automatic control of the control rods, maintaining the reactor in a safe condition at all times.</p> <p>To match reactor power with load demand.</p> <p>To maintain sufficient reactivity in reserve for burnup, long-term transients, startup, shutdown and refueling.</p> <p><u>a) Control Rod Drive</u></p> <p>To allow rapid rod insertion under scram conditions.</p> <p>To operate the rod in small steps so that reactivity and temperature changes are acceptable to fuel and plant.</p> <p>To allow small, rapid rod insertions to control small and short-term transients.</p> <p>To allow rod withdrawal so that startup from hot shutdown conditions is reasonably rapid (e.g., about 1 hr. from determining that a trip is spurious).</p> <p><u>b) C.R.D.M. Electronic Circuitry</u></p> <p>To maintain all C.R.D.M.'s in a state of operability and provide a continuous series of synchronous pulses.</p> <p>To determine the control rod motions required from plant monitor signals and send a command signal to the appropriate C.R.D.M.'s.</p>	<p>Overall reactor control analysis methods are sufficiently developed from current power reactor technology.</p> <p>Westinghouse Magnetic Latch C.R.D.M.</p> <p>Logic and hardware for single module reactors exists from current power reactors.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Data concerning reactivity coefficients at all power levels are required.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Data concerning lifetime and performance when operating in a noble gas with sodium vapor present.</p> <p>Accuracy and reliability data from tests under operating conditions.</p> <p><u>Engineering Data and Techniques</u></p> <p>Investigation of limit cycle stability problems associated with coefficients, deadbands, signal delays, etc.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Development of logic circuits for operating control rods.</p> <p>Development of safety features.</p>	<p>6.2.A</p> <p>6.2.A</p> <p>6.2.A</p> <p>6.2.A</p> <p>6.2.A</p> <p>6.2.A</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Plant Control Systems (Paragraph 14-2.2)			
<p>a) <u>Sodium Pump Speed Control Systems (Primary and Secondary)</u> To control primary and secondary coolant flows during full and partial power operation so that the terminal temperature differences in the plant, IHX's and steam generators do not cause stresses in excess of design stress limitations.</p> <p>To allow individual control and ganged operation of all pumps.</p> <p>To allow rapid flow variation so that large temperature rates do not occur at core inlet causing reactivity accidents.</p> <p>b) <u>Steam Generator Control Systems</u> The sodium flow is controlled by the secondary sodium pump utilizing feedback from the steam generator control system.</p> <p>To control the sodium and feedwater flows so that the load demand is met and the steam pressure and temperature are within design limits for full and partial power operation.</p> <p>To match feedwater temperature on full and partial power operation to that of the sodium to maintain material stresses within design limits.</p> <p>To locate the position of the boiling boundary and insure that boiler dryout and turbine flooding do not occur.</p> <p>To isolate the steam generator on detection of a leak.</p>	<p>Experience is available for multiple loop, two coolant systems from current power reactors (PWR) limited experience of three coolant system plants from existing LMFBR's. Little or no experimental back-up information for current simulation methods is available.</p> <p>State-of-the-Art reference: LMFBR Program Plan, Volume 4.</p> <p>Experience with water/water and gas/water steam generators from current power reactors (PWR, Magnex, AGR). Limited experience with sodium/water steam generators exists.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Overall plant dynamic simulation techniques must be developed and as far as possible verified against actual performance.</p> <p>Definition of operational procedures for individual pumps and ganged pumps under both normal and fault conditions.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Development of control circuitry for a multiple loop, three coolant heat removal systems, including interconnection and ganged control of the pumps.</p> <p>Data regarding the sensitivity of control loops to changes in system variables.</p> <p><u>Engineering Data and Techniques</u></p> <p>Development of adequate steam generator simulation models and the necessary verification of these models.</p> <p>Development of security of control to prevent loss of feedwater flow and define minimum flow required to prevent boiler cryout under various conditions.</p> <p>Development of parallel channel flow stability models which predict flow oscillations.</p> <p>Data concerning the accuracy required from temperature and pressure regulation and the effects of deadband width (using models developed in 1).</p> <p>Determination of effects of thermal transients due to loss of feedwater flow (using models developed in 1).</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Detection of leaks and sodium/water reactions and total isolation of steam generator from secondary and tertiary systems.</p>	<p>6.2.B</p> <p>6.2.B</p> <p>6.2.B</p> <p>6.2.B</p> <p>6.2.B</p> <p>6.2.B</p> <p>6.2.B</p> <p>6.2.B</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Digital Control and Data Logging (Paragraphs 14-2.1, 14-2.2, 14-2.3)			
<p><u>a) Computer</u></p> <p>To monitor sensor information from the plant, calculate significant parameters, measure against pre-established performance criteria and transmit a signal to an analog control circuit.</p> <p>To collect and log the data received from plant sensors.</p> <p>Using data from in-core sensors and activation instruments, maintain an inventory of material in the plant and recommend action for breeding optimization.</p> <p><u>b) Interface and Buffer Store</u></p> <p>To provide accurate analog/digital conversion of sensor signals and store until needed those not required immediately.</p> <p>To provide accurate digital/analog conversion of control signal from the computer to the analog control circuit.</p>	<p>Plans were made for digital control of UTREX, LAMPRE and FFTF.</p> <p>Oldbury (UK) computer alarm analysis.</p> <p>Hinkley Point B (UK) computer data presentation to operators and logging.</p> <p>Rapsodie (France) computer data presentation to operators and logging.</p> <p>Techniques adapted from hybrid computers.</p> <p>State-of-the-Art reference:</p> <p>LMFBR Program Plan, Volume 4.</p> <p>ANL-7380: Proc. Symp. LM Instrumentation and Control.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Development of computer programs for data manipulation, alarm, data logging, and control functions.</p> <p>Development of fuel management programs.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Development of digital equipment systems which allow adequate plant security and safety.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Development of reliable hardware for interface.</p>	<p>6.2.C</p> <p>6.2.C</p> <p>6.2.C</p> <p>6.2.C</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
INSTRUMENTATION AND CONTROL (Section 14) - Reactor Trip System (Paragraph 14-2.1)			
<p>To prevent the plant exceeding its established design limits during normal and abnormal operation.</p>	<p>Reactor trip system hardware has been extensively developed for current thermal reactor systems.</p> <p>The relationship of this equipment to a 1000 MWe LMFBR operational and safety requirements remains to be defined.</p>	<p><u>Engineering Data and Techniques</u></p> <p>The significant transients for use as trip system design basis.</p> <p>Evaluation of safety channel performance and consequent operating characteristics for trip system for the design basis transients.</p> <p>Identification of any necessary further development required.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Development of required sensors circuits and logic to fulfill its safety function.</p>	<p>6.2.D</p> <p>6.2.D</p> <p>6.2.D</p> <p>6.2.D</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR SHIELDING SYSTEM (Section 5) - Radiation Shielding (Paragraph 5-2)			
<p>To:</p> <p>a) Reduce the radiation to cope with manned access requirements throughout the plant (~Irem/quarter).</p> <p>b) Meet material irradiation limits.</p> <p>c) Meet material and components activation limits.</p> <p>d) Reduce the heat flux and nuclear heating to required limits (e.g., radial limit 20 Btu/ft² hr. heat flux on concrete).</p> <p>Reactor shield plug must attenuate shutdown gamma dose rate from radioactive primary sodium in the reactor by a factor of 10⁻⁶ in the hot cell.</p> <p>Sodium in the secondary loop must be shielded sufficiently to yield a specific activity of less than 1 mCi/cc.</p> <p>The containment wall must provide sufficient shielding to attenuate the radiation level to less than 2 mR/hr outside of the containment.</p>	<p>Existing nuclear plants have a lower rating and differing spectra.</p> <p>Serpentine aggregate has been used in shielding in:</p> <p>Fermi: γ shielding only</p> <p>SEFOR: neutron and shielding</p> <p>Principal shielding material for the W Reference LMFBR Plant also.</p> <p>State-of-the-Art reference: LMFBR Program Plan, Volume 10, Volume 2, and Volume 9.</p> <p>[with reservations, particularly regarding secondary sodium activity, and the application of conservatism regarding the feasibility of the use of critical facilities for streaming experiments and regarding structural damage mechanisms.]</p>	<p><u>Basic Data</u></p> <p>Nuclear data (e.g., cross sections) for constituent materials, such as serpentine aggregate, is either unavailable, incomplete or in a difficult-to-use form. Basic cross section experimentation is required.</p> <p>More data on certain physical properties (e.g., thermal conductivity, coefficients of thermal expansion) of some shield materials.</p> <p><u>Engineering Data and Techniques</u></p> <p>Data on the dependence of certain properties (e.g., material strengths) on temperature and on neutron and gamma irradiation requires determination.</p> <p>In the event of abnormal conditions in which sodium is freely in contact with shielding materials, its chemical interaction with hydrous constituents (e.g., serpentine) is unknown.</p> <p>A suitable processing code for converting basic nuclear data into a more usable form is required.</p> <p>For operation on the CDC-6600, a two dimensional multigroup removal-diffusion code for both neutron and gamma flux calculations, together with a complete library suitable for use with the code is needed.</p> <p>A transport theory code for final design assessment (one and two dimensions). Such codes are required for both neutron and gamma flux calculations together with a complete library for use in each case.</p> <p>For a shielding design, a Monte Carlo code (such as O5R), together with its library, is needed. Immediate use would be for streaming calculations and secondary sodium activity determination.</p> <p>Data on the release of fission products from the vented fuel to the coolant, the rate of release and the subsequent transportation within the primary loop requires study.</p> <p>Experimental data to verify methods used in digital calculations.</p>	<p>8.1.A</p> <p>8.1.B</p> <p>3.2.C</p> <p>8.1.B</p> <p>8.1.B</p> <p>8.1.C</p> <p>8.1.D</p> <p>8.1.E</p> <p>8.1.F</p> <p>8.1.I</p> <p>8.1.H</p> <p>8.1.G</p> <p>8.1.I</p> <p>8.1.J</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
REACTOR SHIELDING SYSTEM (Section 5) - Blast Shielding (Paragraph 5-2)			
<p>To absorb shock loading following a core meltdown and compaction accident to avoid shock damage to primary containment structure.</p> <p>The shield doubles as a radiation shield.</p> <p>Shock energy from 2000 lbs. TNT. Maximum circumferential elongation allowed 30%.</p> <p>Design Base Accident: Core meltdown and compaction.</p>	<p><u>SEFOR</u></p> <p>4 rows SS tubes filled with Boron Carbide plus 7 rows of carbon steel tubes filled with serpentine rock. Then a 6 3/4" thick annular serpentine container plus 1 1/2" carbon steel liners plus 2" thickness of steel in 5 concentric steel shells.</p> <p>Shock energy from 200 lbs. TNT. Maximum circumferential elongation allowed 20%.</p> <p>Top: Plug adequately constrained by bolts.</p> <p>Design Base Accident: Core meltdown and compaction.</p> <p>State-of-the Art reference: LMFBR Program Plan, Volume 10.</p>	<p><u>Engineering Data and Techniques</u></p> <p>Shock energy data resulting from core meltdown and compaction accident and model for distribution of this energy.</p> <p>Material properties (e.g., strain and stress strengths and densities of serpentine and steel for deformation calculations at high deformation rates.</p> <p>Confirmation of the effects of a TNT explosion on a serpentine steel shield by experimental methods.</p> <p>Modeling to take end effects into account in the design of radial blast shielding.</p>	<p>8.3.H 8.3.I</p> <p>8.2.A</p> <p>8.2.B</p> <p>8.2.C</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
SAFETY SYSTEMS (Section 16) - Containment Building (Paragraph 16-2.4.1)			
<p>Internal pressure rating: 30 psig.</p> <p>Volume of approximately 10^6 ft³.</p> <p>Design Basis Accidents: a) Sodium Fire b) Meltdown and Compaction</p>	<p>Such containments have been built:</p> <p><u>Beznau-Dottingen</u> (Swiss) (350 MWe PWR)</p> <p>3 ft. concrete shell cylindrical above grade with larger radius domed cap.</p> <p>Internal pressure rating: 38 psig.</p> <p>Containment volume: $1.9 \cdot 10^6$ ft³ and other PWR system containments.</p> <p>Design Base Accident: Depressurization.</p> <p><u>Sodium Fire</u></p> <p>State-of-the-Art reference: LMFBR Program Plan, Volume 10.</p> <p><u>Meltdown and Compaction</u></p> <p>State-of-the-Art reference: LMFBR Program Plan, Volume 10.</p>	<p><u>Engineering Data and Techniques</u></p> <p><u>Sodium Fire</u></p> <p>Confirmation of published data such as flame temperatures, burning rates, radiation emissivities, and heat transfer coefficients by experimental methods.</p> <p>Sodium fire data from the development of an adequate model of a Sodium Fire including possible self-limitation of the fire by blanketing, energy release, temperature distribution within the sodium, accurate representation of flame and air-combustion products mixture.</p> <p>Experimental data on burning rates for sodium spray fires including model for burning mechanism.</p> <p>Calculation of burning rates for sodium fires under explosive ejection conditions including model for burning mechanism.</p> <p>Data on the aerosol spread of radioactivity following a spray fire.</p> <p>Data for the development of engineered safeguards to inhibit and terminate any sodium fire, e.g., restrictive vault geometry to prevent access to large volumes of air, preferential sodium flow to safe areas, etc.</p> <p><u>Meltdown and Compaction</u></p> <p>Rates and amounts of activity changes in a core reassembly both when the core reassembly is due to gravitational collapse and shock compaction. Model for fuel collapse mechanism.</p> <p>Data on mode of collapse from induced fuel failure and meltdown experiments.</p>	<p>8.3.A</p> <p>8.3.A</p> <p>8.3.B</p> <p>8.3.B</p> <p>8.3.C</p> <p>8.3.D</p> <p>8.3.E</p> <p>8.3.E</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
SAFETY SYSTEMS (Section 16) - Containment Building (Paragraph 16-2.4.1)(Continued)			
		<p>Data on physical, chemical and nuclear properties of core materials under extreme accident conditions. The equation-of-state for core materials (to determine pressure feedback), the Doppler effect for fuel materials (to determine nuclear feedback) require more exact specification.</p> <p>Data for autocatalytic effects in expansion mode to gauge their importance.</p> <p>Further data on the partition of energy following the dispersion of core materials. The energy required for missile propulsion and sodium vaporization as well as initial shocks requires definition. Model for calculation of transient conditions within the containment.</p> <p>Possible experimental data for the confirmation of dispersal model.</p>	<p>8.3.F 8.3.G</p> <p>8.3.H</p> <p>8.3.I</p> <p>8.3.1</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
SAFETY SYSTEMS (Section 16) - Protection Against Loss of Coolant (Paragraph 16-2.2)			
<p><u>a) Check Valves</u></p> <p>Primary circuit check valves located close to inlet nozzles to the vessel.</p> <p><u>b) Siphon Bleakers</u></p> <p>Siphon check valves located at upper end of inner vessel barrel.</p> <p><u>c) Pipework Secondary Containment</u></p> <p>Primary pipework between pump and vessel inlet designed with secondary containment.</p> <p>Auxiliary cooling system pipework concrete encased.</p>	<p>Fermi: Balanced disc valves at pump discharge positions.</p> <p>Siphon breakers are included in the Fermi and FARET coolant loops.</p> <p>Fermi: All primary pipework with secondary containment if below reactor level datum.</p> <p>FARET: The whole of auxiliary cooling pipework has secondary containment.</p>	<p><u>Engineering Data and Techniques</u></p> <p><u>Pipe Rupture</u></p> <p>Hydraulic data from a pipe rupture model to predict flow patterns within the reactor system following a primary pipe break of any size up to a double ended circumferential break.</p> <p>Calculated data for the comparison of consequences of pipe ruptures occurring prior to and following the main pump to confirm necessity for the secondary containment of pipework up to the vessel inlet following the pump.</p> <p>Data on pipe failure modes under scaled operating conditions of the plant, including the failure modes of piping within piping.</p> <p>Steady state thermohydraulic data for a model and code to calculate conditions of the plant primary loops, paying particular attention to the action of valves and flow valves in each loop.</p> <p>Data on erosion of valve seatings in sodium to check feasibility of siphon breakers allowing percentage bypass flow in normal operation.</p> <p><u>Experimental Verification and Operational Analysis</u></p> <p>Experimental data and confirmation of pipe rupture model employing scaled simple flow loops.</p> <p>Experimental data for shocks to primary system following check valve closure.</p> <p>Experimental data for the confirmation of operation and efficiency of chosen siphon breaker valve system under high pressure differentials.</p>	<p>8.4.A</p> <p>8.4.B</p> <p>8.4.B</p> <p>8.4.E</p> <p>8.4.F</p> <p>8.F.C</p> <p>8.4.D</p> <p>8.4.F</p>

CRITICAL REFERENCE PLANT FUNCTIONAL REQUIREMENTS	CURRENT TECHNOLOGY	INFORMATION NEEDS TO IMPLEMENT REFERENCE PLANT	R & D TASKS (SECTION III)
SAFETY SYSTEMS (Section 16) - Protection Against Secondary Criticality (Paragraph 16-2.3)			
<u>a) Terminal Cooling System</u> (Not part of reference design.) <u>b) Core Dispersal System</u> (Not part of reference design.) Device to disperse core into subcritical formation or otherwise prevent secondary criticality.	SEFOR: ZrO_2 coated SS cone below vessel plus additional conical shield support on shield plug plus 4 ft. deep SS and calcium silicate insulation lined material catch tank with a 4 ft. high cone in the center. Dounreay: Multiple cones in and out of vessel leading to 24 catchment pipes in foundations. Fermi: Core inlet dispersal cone plus Zr lined tray and steel baffles around a further cone, plus, outside the vessel, a borated graphite-filled steel crucible capable of containing molten uranium. State-of-the-Art reference: LMFBR Program Plan, Volume 10.	<u>Engineering Data and Techniques</u> Data on fuel failure details, the movement of molten fuel and the mechanisms of core collapse. Data for the use of refractories to cladding dispersal devices (e.g., ZrO_2 , Silicates). Data to assess whether a solid or liquid phase heat sink would be more suitable (e.g., solid heat sinks or convection paths or liquid convection cooling). <u>Experimental Verification and Operational Analysis</u> Experimental information relating to the final distribution of the molten fuel as a result of passage past the dispersal system.	8.5.A 8.5.B 8.5.D 8.5.C
<u>c) Radiological Protection, Aerosol Settling Device</u> (Not part of reference design.)	State-of-the-Art reference: LMFBR Program Plan, Volume 10. Conf. 660904. September 1966 Air Cleaning Conference.	<u>Experimental Verification and Operational Analysis</u> Experimental data to review different aerosol protection device designs (e.g., high volume filtering system, electrostatic deposition, sonic deposition systems, thermal precipitators, etc.).	8.6.A 8.6.B

SECTION III

PROGRAM DESCRIPTIONS

1. PROGRAM CONCEPT SUMMARY

The Westinghouse concept of the LMFBR program basically coincides with the LMFBR Program Plan summarized in Volume 1. In Figure 1.1-1, a simplified overview of this concept is presented.

The sustaining base consists of the Base Technology Programs. These programs are largely AEC supported. They provide for development of technical data having broad application and not tied directly to plant projects. The base includes the provision and operation of National Facilities which represent capital investments that could not be amortized by industrial firms on the basis of anticipated returns of the one firm alone. The Fast Flux Test Facility (FFTF) and Sodium Pump Test Facility (SPTF) are examples of such facilities.

These Base Technology Programs will sustain several Demonstration Plant Programs and subsequent extrapolations to one or more Reference Plants. With the achievement of marginal economic viability as represented by commitment of one or more reference plants with minimum or no Federal support, the breadth of Base Technology Programs will diminish, as it has in the maturing, light-water reactor economy. It will continue, however, to provide limited support of evolutionary development of commercial plants through the medium of unique National Facilities and associated programs.

Many of the Base Technology Programs have been initiated. Their direction in the future will be guided by information needs developed through AEC sponsored Target Plant Studies, such as this study, and by means of privately funded Demonstration Plant Design Studies.

Figure 1.1-1. LMFBR Program Concept

The technical and economic risk involved in moving directly from the Base Technology Program to the Reference Plant is considered to be too large to be assumed by the reactor manufacturer and utility. It is considered necessary, therefore, to demonstrate as many features of the Reference Plant as feasible and prudent in a smaller Demonstration Plant whose size and technical features represent a manageable risk. Although the Demonstration Plant can be justified on that basis alone, it also serves as an efficient vehicle for proof testing many systems and components. The Demonstration Plant will provide data that can be developed in no other way, such as reactivity lifetime, statistically significant fuel irradiation data, systems response, and large-size heat exchanger performance.

In many technical areas, the Demonstration Plant Program can provide all information needed for the Reference Plant. Fuel handling machines are an example of a component whose development may be considered complete with operational analysis of Demonstration Plant operation. Other areas will require additional development to implement the Reference Plant; such examples are extrapolation of large sodium pump size and improvement of fuel performance.

Although the development programs assume success in attaining Reference Plant goals with the first 1000 MWe or equivalent plant, the probability of this achievement is less than unity. As the program develops, it may be necessary to progress through an intermediate size Demonstration Plant or a Reference Plant with some features short of the goal. Such eventualities are analyzed in the Value Analysis Section of this report.

Finally, Reference Plant features will be subject to evolutionary development and cost reductions in successive plants. Most of such improvement will be sponsored by industry, supplemented, particularly fuel development and reprocessing, by the National Laboratories and National Facilities.

2. PLANT DESIGN STUDIES

As stated in Volume 2 of the LMFBR Program Plan, "The objectives of the U.S. Atomic Energy Commission's Liquid Metal Fast Breeder Reactor Program are aimed at the development of power plants for the utility environment ... The attainment of a satisfactory LMFBR Plant is the first and foremost goal of the Program."

Westinghouse concurs in the General Approach used in the Program Plan, that is:

- a. Studies of target plants to guide research and development and the features of Demonstration Plants, the successful operation of which would provide confidence for extrapolation to larger sizes
- b. Construction of several first generation Demonstration Plants by joint AEC-Electric Utility Industry-Reactor Designer/Manufacturer partnerships
- c. Operations analysis of Demonstration Plants with continued development and additional Demonstration Plants of intermediate size if required to meet program goals.

To meet these objectives, a number of closely related and integrated programs are and will be performed utilizing partnerships of the AEC, members of the Electric Utility Industry, and Reactor Designer/Manufacturer. Some of these programs will run concurrently while others will be initiated as their predecessors are completed. The most important factors in this program concept are the need for a common goal and the unimpeded flow of information from program to program. Without both of these qualities, the objectives of the LMFBR Program will be very difficult, if not impossible, to obtain.

The three basic types of program involved in the overall concept are Conceptual Design Studies, Base Technology Programs, and Plant Design and Operation. The relationships between these are shown in Figure 1.1-1. The remainder of this section will discuss the Conceptual Design Studies and the Demonstration Plant Program, and how both will contribute toward the development of the information required to meet the LMFBR Program Objectives. The Base Technology Programs will be discussed in the subsequent sections of this document.

2.1 Conceptual Design Studies

Early in the evolution of the overall LMFBR Program, it became apparent that the goals could best be achieved if target plants were conceived and studied prior to embarking on the various phases of the research and development programs needed to accomplish this end. The research and development programs, which were to involve a large expenditure of time, manpower, and material, were to be directed toward solving problems oriented to the target plants.

The AEC, through the Argonne National Laboratory, has funded a number of conceptual studies of the target plants called "1000 MWe Fast Breeder Reactor Follow-on Studies". In addition, it is recognized in Volume 2 of the LMFBR Program Plan that a number of studies of certain aspects of the target plants must also be undertaken to fully evaluate its ultimate safety, reliability, and economic viability.

2.1.1 1000 MWe Fast Breeder Reactor Follow-on Study

2.1.1.1 Objective

Westinghouse is one of five industrial contractors performing 1000-MWe LMFBR Follow-on Studies for the Argonne National Laboratory. The primary objective of these studies is to provide information which will be of assistance to the AEC in the furtherance of the National LMFBR Program Plan. This broad objective is used to guide the following specific objectives of the study:

- a. To conceptually design a nuclear steam supply system and energy conversion system which could be committed to sale in 1980
- b. To optimize this conceptual design for minimum power cost
- c. To perform a number of parametric analyses
- d. To identify data and information, not presently available, which must be obtained to permit the safe and economic construction of an LMFBR Plant based on this design
- e. To recommend research and development programs needed to supply the data and information
- f. To assess the value of such research and development programs.

2.1.1.2 Program Summary

To accomplish these objectives, the study has been divided into four tasks. The bulk of Task I is devoted to consideration of alternate design concepts and trade-off studies between concepts. Task II entails the conceptual design of the plant resulting from the choices made in Task I. The conceptual design from Task II is the basis for parametric analyses, a plant availability analysis, and a redesign based on power cost optimization in Task III. In Task IV the programs for research and development which are required to bring into operation a plant based on the final design are recommended and evaluated (and presented in this document).

2.1.1.3 Relation to LMFBR Program Plan

The 1000 MWe Follow-on Studies are an integral part of the AEC LMFBR Program Plan. The results of these studies will guide research and development programs and the selection of features of the Demonstration Plants.

The studies depict the target plants which have been established prior to performance of the R&D and Demonstration Plant Programs. Therefore, these studies should be periodically updated to reflect the results obtained from all related programs. This aspect of the program has been defined in Volume 2 of the LMFBR Program Plan.

2.1.1.4 Major Test Facility Requirements

None required for this study.

2.1.1.5 Program Schedule

These studies began in calendar year 1967 and will be completed by the end of fiscal year 1969.

Updating the study is not shown on the schedule, but it is anticipated that a major updating would be in order about 1972 when significant R&D results should be available and while detailed design of the Demonstration Plant is in progress. Another major updating might be expected as a result of input from the Demonstration Plant Post-Construction R&D Program.

2.1.2 Other Design Studies

Volume 2 of the LMFBR Program Plan defines four Task Areas under the technical program in which studies must be performed to evaluate concepts, designs, and features for the target plants. These areas are:

Task Area 1 - Design Development Studies

Task Area 2 - General Plant Studies

Task Area 3 - Demonstration Plant Program

Task Area 4 - Other Plant Design Activities

The 1000 MWe Follow-on Studies, a part of Task Area 1, are described above. The Demonstration Plant Program, Task Area 3, will be discussed in Section 2.2 of this report.

There are twenty-one other programs in the four Task Areas which are quite comprehensive as a group. Westinghouse concurs that these programs (Table 2.1-1) are required to fully examine the potential problem areas in the LMFBR Target Plant Design, and that the addition of a program for the design and evaluation of the Plant Auxiliary Systems is required (added to the table as Subtask 1.11). This Subtask is the only one which will be described below, the others being adequately described in Volume 2 of the LMFBR Program Plan.

TABLE 2.1-1

LIST OF DESIGN STUDIES

Ref: LMFBR Program Plan, Volume 2, Plant Design

Subtask No.	Subtask Title	Organization ^a
Task Area 1: Design Development Studies		
1.1	1000 MWe Follow-on Studies	RD, NL ^b
1.2	Study of Pool (Submerged) Primary System	RD
1.3	Study of Loop (Piped) Primary System	RD
1.4	Plant Safety Analyses Study	RD
1.5	Vented-Fuel Effects on Plant Design	RD
1.6	Refueling System Study	RD
1.7	Dynamic Analysis and Control Study	RD
1.8	Engineering Safety Systems: Containment and Shielding Study	RD
1.9	Plant Maintainability Study	RD, U
1.10	Plant Design Temperature Study	RD, NL, C
1.11	Plant Auxiliary Systems Design Study	RD
Task Area 2: General Plant Studies		
2.1	Plant Siting Study	U, AE, C, RD
2.2	Development and Evaluation of Plant Economic Factors	AE, RD
2.3	Specifications and Standards	AE, RD, CV, U
2.4	A User's Assessment of LMFBR Plants	U
2.5	Investigation of More Compatible Secondary/Steam Energy Transport Systems	RD
2.6	Elimination of Secondary System	RD, NL

TABLE 2.1-1 (Continued)

Subtask No.	Subtask Title	Organization ^a
2.7	Secondary System Design	RD, AE
2.8	On-Line Refueling Study	NL
2.9	Advanced Uses of LMFBRs	NL, RD
Task Area 4: Other Plant Design Activities		
4.1	Compilation, Assessment and Dissemination of Information	RD, NL
4.2	Identification of Functional Requirements	NL, RD
4.3	Program Assessment	

a. Type of implementing organization

b. Program management

RD - Reactor Designer

NL - National Laboratory

U - Utility

C - Consultant

CV - Component Vendor

AE - Architect-Engineer

Plant Auxiliary Systems Design Study

Plant Auxiliary Systems for an LMFBR can consist of the following systems:

- a. Sodium Service Systems
- b. NaK Service Systems
- c. Inert Cover Gas System
- d. Compartment Environmental Control
- e. Containment Inerting System
- f. Gaseous Radioactive Waste Disposal System
- g. Liquid Radioactive Waste Disposal System
- h. Solid Radioactive Waste Disposal System
- i. Equipment Cleanup and Decontamination Systems
- j. Plant Electrical Systems

Each of the first five systems may handle large quantities of radioactive material while maintaining high standards of purity requirements of the working fluid. The next four systems will also handle large quantities of radioactive material. All of these systems will consist of many components each designed such that the overall system will exhibit a high degree of reliability, and will be designed to the highest safety standards. These systems will require a great deal of radiation shielding surrounding them and many will be located in the containment structure where space is at a premium for economic considerations. Therefore, these systems must be efficiently and reliably designed, packaged, and operated. The plant electrical system requires the highest attainable reliability to serve the needs of the plant as well as meet the protective system and engineered safeguards requirements.

2.1.2.1 Objective

The objective of this task is to develop a thorough understanding of the engineering requirements of the plant auxiliary systems and to relate these to plant operation and safety, as well as physical size and arrangement. Specific objectives will include:

- a. An understanding of the technical and economic considerations of designing, constructing, and operating such systems

- b. An evaluation of the safety and reliability of these systems to prevent the limitation or control of plant availability
- c. Identification of R&D requirements for components, subsystems and complete systems, and assessment of the value of these requirements

2.1.2.2 Program Summary

To accomplish the above stated objectives, this program is divided into three tasks. Task I will be utilized to evaluate the designs of the target plant auxiliary systems and their subsystems and components to achieve the best design from technical and economic considerations. Under Task II, reliability studies of these systems will be performed to assure compatibility with the target plant availability. Task III will be utilized to define the R&D programs required to evaluate the performance and reliability of these systems and to evaluate the relative needs of those requirements.

2.1.2.3 Task Descriptions

Task I - Design Evaluation

The objective of this task is to evaluate the design of the target plant auxiliary systems and their subsystems and components to obtain the best design from a technical and economical standpoint. With this objective in mind, the design of the auxiliary system will be reviewed with respect to system functions and design requirements which will be established for this purpose. The system interfaces will be defined. The performance of the systems will be assessed as to their compatibility and influence on other plant systems. Promising alternate designs will be evaluated. Certain areas which are especially important to plant availability include efficient sodium and gas purification (i.e., oxygen, nitrogen, fission products, etc.) systems, leak integrity of various environmental control systems, handling of radioactive waste disposal products, decontamination of equipment, and auxiliary electrical systems integrity.

Task II - Reliability Studies

The objective of this task is to determine the reliability of the auxiliary systems to assure compatibility with the target plant availability and operation. Because some of the auxiliary systems are crucial to plant operation and safety, their influence and designs must be assessed as to their reliability. Reliability studies will be performed to determine this influence for each auxiliary system. These results will be assessed with respect to compatibility and influence with other plant systems, operation and safety. Each component and subsystem must be evaluated to fully define the reliability of these systems. Such important subsystems and components as the purification subsystems, including fission product removal if necessary, vapor traps, heating and cooling for environmental control systems, filtering units, systems for collecting and disposing of radioactive wastes, and auxiliary electrical systems are influential in determining plant operation and availability.

Involved in this task are:

- a. Review of all operations of the auxiliary system to determine component failures that could arise, their effect, and ways for alleviation and maintenance.
- b. Performance of failure analyses on each vital component.
- c. Determination of the needs for maintenance, abnormal situations, and malfunction, utilizing the results of the above analyses.
- d. Recommendation of design revision.

Task III - Definition of R&D Program

The objective of this task is to define the R&D program required to evaluate the performance and reliability of these systems and to evaluate the relative needs of these requirements. The R&D areas will be identified and programs will be defined to successfully develop these areas for the target plant. These programs will include definition of required development area, scope of effort required, brief description of the base of current technology, information needs, and program description

including schedule of manpower and time, facility requirements, and cost estimate. An assessment of probability of success of these ventures should also be made.

2.1.3 Demonstration Plant Conceptual Studies

The Demonstration Plant will be one of the major tools used to achieve safe, reliable, and economic Target Plants. Therefore, it must be conceived, designed, constructed, and operated with the thought in mind that it is a test of ideas, concepts, materials, systems, and components. The results of testing the Demonstration Plant will provide the information required to design the Target Plants.

2.1.3.1 Objective

The objective of these studies is to determine the optimum size, cost, technical problem areas, and financial uncertainties associated with a Fast Breeder Reactor Demonstration Plant which could be committed for construction in 1970.

Specifically, these objectives include:

- a. A conceptual design of a plant which prototypes as many major concepts of the Target Plant as are required and are possible.
- b. An assessment of the financial risks associated with the plant.
- c. An evaluation of the plant safeguards to ascertain its safety and licensability.
- d. Identification of R&D requirements both for this plant and for subsequent Target Plants and an assessment of the relative needs of these programs.

2.1.3.2 Program Summary

The objectives listed above will be met through: conceptual design of a 200-400 MWe Sodium-cooled FBR Demonstration Plant; development of capital, fuel cycle, operation, and maintenance costs; analysis of the technical and financial uncertainties associated with construction and operation of a Demonstration Plant; and safeguard analyses.

The conceptual design for a 200-400 MWe plant will be developed to a depth sufficient to provide a basis for: obtaining AEC support through the Power Reactor Demonstration Program; evaluating the licensability of the plant; assessing the financial uncertainties; and identifying the technical uncertainties which would require resolution prior to a plant construction commitment in 1970.

The conceptual design effort will be coordinated with a parallel program, funded by the AEC, for a conceptual design and analysis of a 1000 MWe LMFBR. The 1000 MWe plant study, which will lead to an economical commercial plant, will guide design of the 200-400 MWe plant through concept selection and definition of features which will be then prototyped.

A safeguards analysis of the WASB plant will be performed early in the program in order to develop realistic criteria for the 200-400 MWe plant. A safeguards analysis will be performed in sufficient depth to obtain a preliminary review by the regulatory authorities.

The program will be conducted through performance of the following major task:

100	Project Management
200	Concept Selection
300	Conceptual Design and Cost Estimate
400	Cost Analysis
500	Safeguards Analyses

2.1.3.3 Task Descriptions

Task 100 - Project Management

This task provides for technical planning and direction of the project, coordination with related Westinghouse utility and AEC-sponsored programs, liaison with Electric Utility Industry sponsors, project control, and technical progress and financial reporting.

Task 200 - Concept Selection

This task provides for selection of the Demonstration Plant design concepts to best meet the requirement for prototyping the features of the

objective large commercial plant, within prudent limitations of technical and financial risk.

The decisions which were made during the Task I phase of the "1000 MWe FBR Follow-on Study" were reviewed to determine those characteristics which should be prototyped in the Demonstration Plant. Complete correspondence of Target and Demonstration Plant features was considered neither necessary nor prudent. Factors which influence selection of features not identically prototypical include:

- a. Anticipated differences between 1970 and 1980 technology
- b. The smaller plant capacity
- c. Relaxation in economic objectives in the Demonstration Plant in favor of safety and reliability
- d. Scaling factors in component size between demonstrated technology and Target Plant equipment
- e. Opportunity for fuel cycle cost improvement over Demonstration Plant life through introduction of high performance fuel after plant reliability has been demonstrated.

Results of the concept selection were reported in WARD-5019 presented at the ANS Annual Meeting, July 10, 1968.

Task 300 - Conceptual Design and Cost Estimate

This task provides for the conceptual design of the Demonstration Plant and development of capital and fuel cycle costs. The design will be based upon the concept selections made in Task 200, revision and extrapolation of the 200 MWe WASB design, and the parallel large plant conceptual design. This task is in progress at Westinghouse.

The design will be developed in sufficient detail to achieve the following:

- a. Provide a basis for initiation of preliminary (Title I) design
- b. Provide system descriptions suitable for use as a basis for a plant contract with the owning utility and a proposal to the AEC for a Power Reactor Demonstration Program Project

- c. Provide a capital and operating cost estimate, including estimates for construction contingencies and provision for unusual maintenance requirements
- d. Develop fuel cycle cost estimates for the first three complete core loadings and for the equilibrium fuel cycle
- e. Permit performance of a safeguards analysis
- f. Identify design uncertainties to provide guidance to the supporting R&D program.

Task 400 - Cost Analysis

This task will be utilized to develop a recommendation for the optimum Demonstration Plant size through a comprehensive analysis of costs and financial risks associated with a Demonstration Plant Project. Elements of cost and risk to be considered include:

- a. Capital cost and contingency
- b. Operation and maintenance cost and contingency for unusual maintenance
- c. Fuel cost and contingency
- d. Power outage

Task 500 - Safeguards Analysis

This task is being utilized to develop realistic safeguards criteria for the Demonstration Plant and perform sufficient safeguards analysis to obtain a preliminary evaluation of the licensability of the plant from the regulatory authorities. Two safeguards analyses will be made, one for the 200 MWe WASB plant and the other for the 200-400 Demonstration Plant.

To develop initial criteria for the Demonstration Plant design, selected accidents analyses were performed, utilizing the WASB as a base of reference. Emphasis was placed on establishing reactivity coefficient limits and containment criteria.

The above analyses will be revised and extended to develop a safeguards analysis of the Demonstration Plant at a real site. A report will be prepared suitable for an informal review by the regulatory authorities, including the AEC Regulatory Staff and the Advisory Committee for Reactor Safeguards.

2.1.3.4 Relation to LMFBR Program Plan

The performance of Demonstration Plant Conceptual Designs was not discussed in the Program Plan. However, the intent that the Demonstration Plant follow a course as outlined above is clear in the Program Plan and this task fulfills that intention.

2.1.3.5 Major Test Facility Requirements

Many R&D programs and the associated facilities will be defined as a result of this study, but none are required to perform the study.

2.1.3.6 Program Schedule

A simplified schedule for "The Westinghouse Conceptual Design Study" is illustrated in Figure 2.1-1 with the major milestones listed on the page following the figure. The AEC-sponsored "1000 MWe Follow-on Study" is also included to show its relationship to this program.

MAJOR MILESTONE KEY TO FIGURE 2.1-1

Large Plant Conceptual Design Study

1. Reactor concepts established (module size, nuclear and T-H characteristics)
2. Plant concepts established (fuel handling, number of loops, primary system arrangement)
3. Conceptual design engineering complete
4. Parametric, economic and safety studies complete
5. Program complete

Demonstration Plant Conceptual Design Study

1. Detailed project schedule complete
2. Start analysis of prototyping features
3. Start conceptual design
4. Safety criteria developed for demonstration plant design
5. Complete WASB safeguards analysis report
6. Complete identification of prototyping features
7. Start demonstration plant safeguards analysis
8. Start cost analysis
9. Conceptual design freeze
10. Preliminary draft of design description
11. Complete design description (draft) and issue cost estimates
12. Issue design description report
13. Complete engineering - demonstration plant safeguards analysis
14. Complete financial analysis
15. Issue demonstration plant safeguards analysis report
16. Design freeze for oxide plant
17. Preliminary draft of oxide design description
18. Complete oxide design description (draft)
19. Issue oxide plant design description

Figure 2.1-1. Simplified Schedule of Joint Westinghouse-Electric Utility Demonstration Plant (Conceptual Design Study)

2.2 Demonstration Plant Program

The Demonstration Plant Program, as described in Task Area 3 in Volume 2 of the LMFBR Program Plan, plays a key role in the development program leading to the Target LMFBR Plants. The execution of the Demonstration Plant Program is described in this section.

2.2.1 Pre-Construction R&D Program

The Pre-Construction R&D Program is utilized to evaluate the probability of success in all key technical areas related to the Demonstration Plant Design. In planning for the necessary and sufficient tasks to be included in this R&D program, one must recognize that the Demonstration Plant Project is one of several programs required to develop a sound base of technology for the long-range LMFBR program. Other prototype programs, intermediate Demonstration Plant Projects, and continued AEC Base Technology Programs are essential to meeting the long-range goal. The immediate requirement is to design, construct and operate a prototype fast reactor. The research and development is a supporting activity.

The Pre-Construction R&D program began at Westinghouse in 1966 and is expected to culminate in a firm commitment for the Demonstration Plant construction in mid-1970. Some of the R&D tasks of this program have long-range application in addition to meeting prototype plant program needs. Other R&D tasks are directed specifically toward providing a basis for design of the prototype plant and do not have a general long-range application.

Planning of this program has emphasized identification of problems which limit technical feasibility and/or schedule. With such identification, resources may be concentrated on these problems early in the program. The five most critical areas from this viewpoint are:

1. Fuel Development
2. Reactor Safety
3. Sodium-Steam Generator Development
4. Fuel Handling Systems Development
5. Fission Product Management

Fuel development is critical both from technical feasibility and scheduling problems. The use of carbide fuel in vented elements is considered necessary to meet doubling time objectives, but the stability and compatibility of the proposed fuel has not been demonstrated. Extensive irradiation requirements and limited fast irradiation space combine to create schedule limitations.

Reactor safety will be a critical development problem if in attaining suitable reactivity coefficients associated with sodium, core configurations designs are required which fail to meet economic objectives.

Technical problems in steam generator tube materials and fabrication, combined with the long fabrication lead times, make sodium-steam generator development critical.

Fuel handling systems development may not present problems of basic technical feasibility. The influence of the fuel handling system on capital cost and plant availability, and the lead times associated with meaningful concept and proof tests, however, combine to make still another critical area.

Vented fuel used to obtain high burnups, essential in meeting economic objectives, requires development of methods for routinely managing fission products in coolant and cover gas. Fission product management also presents problems in technical feasibility and scheduling of development and proof tests.

The program, as now planned, concentrates a large percentage of resources in the five critical development areas discussed above. Westinghouse Nuclear Energy Systems has successfully utilized this approach -- concentration on critical development areas -- in other plant-associated research and development programs. Two of these are the R&D programs for Yankee and the San Onofre plants.

2.2.2 Title I and Title II Plant Design

It is anticipated that this work will be performed as part of an AEC-Westinghouse-Utility, Power Reactor Demonstration Project. Title I (Preliminary Design) will start with the conceptual design (previously discussed) and factor-in pertinent results of the R&D programs to date. As the design

progresses, more and more results of R&D programs will serve as additional input. Design problems that can only be solved by additional small R&D program may arise. It is not anticipated, however, that major concepts will prove unworkable in this phase of the design due to the intense evaluation performed in the concept evaluation and selection tasks.

Title II design (Final or Detailed Design) will be performed following Title I design and should take approximately two years. During this phase many of the results of the R&D program will be available as input. Larger scale, but not full scale, tests of many components and systems will be operational and results of these tests also will be input to the design.

The schedule for both Title I and II Design and their relation of the R&D program is shown in Figure 2.2-1.

2.2.3 Plant Construction

During the Demonstration Plant construction phase, proof tests of many of the critical components will be performed. These tests will cover both steady state and transient operation of these components under reactor-type conditions where possible to insure a high degree of reliability when they are in a plant operation.

2.2.4 Post Construction R&D Program

This program will be performed on the Demonstration Plant, systems, sub-systems and components to determine their actual capability under plant conditions. Information obtained will be compared with design calculations to determine the ability to predict performance.

Some major areas that are likely to be investigated include:

- a. Reactor physics
- b. Plant and particularly core thermal and hydraulic characteristics
- c. Fuel performance
- d. Plant chemistry and control thereof
- e. Performance of pumps, heat exchangers, and steam generators
- f. Performance of the fuel handling equipment

Figure 2.2-1. Demonstration Plant Program

- g. Plant transient analysis
- h. Radiation analyses and surveys

The information obtained from these programs will be directly applicable to the Target Plant design.

It is anticipated that this program will last for about five years after reactor criticality.

The early cores of the Demonstration Plant will be fueled with mixed U-Pu oxide. As the technology of the mixed carbide fuel progresses sufficiently to warrant fabrication and testing of complete assemblies, first, individual carbide fuel assemblies will be installed and tested in the Demonstration Plant followed by partial and eventually, full core loadings.

2.2.5 Plant Operational Analysis

During and following the Post Construction R&D Program, operational analyses will be performed for the Demonstration Plant. The plant will be evaluated at steady state operation, its transient behavior will be analyzed, and the stretch capability with advanced fuel will be evaluated. All of this will lead to an economic evaluation to determine the power generating costs that might be expected in the larger LMFBRs. These costs should be maximum values anticipated for subsequent reactors because results from the Demonstration Plant Program and the continuing R&D programs will be used in the design of these plants to minimize costs and risks and maximize safety reliability and power generation.

3. FUEL AND MATERIALS

3.1 Fuel Rod Development

3.1.1 Objective

To develop fuel materials, cladding materials, and a fuel rod concept capable of operating at the high ratings to the high burnups ($\sim 100,000$ MWD/T) demanded in the reference core design, with a low probability of failure. Overpower conditions of operation should not cause gross rod failure of the type that would release a large amount of fuel into the coolant or would cause failure of adjacent rods.

3.1.2 Program Summary

Fuel rod development is divided into three major areas:

- a. Fuel properties evaluation
- b. Cladding properties evaluation
- c. Fuel rod performance characterization

Nine separate tasks are identified from these areas. These tasks and their inter-relation, in terms of data output are illustrated in Figure 3.1-1.

The end-point of this development program is a prototype fuel rod design. This design also must be evaluated as a subassembly component.

The specific objectives of the nine fuel rod development tasks are detailed in the next section.

3.1.3 Task Descriptions

Task 3.1.A - Evaluation of Fuel Fabrication Processes

The objective of this task is to develop a fabrication process for carbide fuel material having the following characteristics:

- a. High quality, reproducible product
- b. High throughput, preferably from a continuous process
- b. Low cost.

Figure 3.1-1. Model of Fuel Rod Development Process

Initially, fabrication processes will be developed for Cr-modified, near-stoichiometric and hyperstoichiometric fuels. Representative fuels of the three types, perhaps produced by alternate processes will be evaluated in Tasks 3.1.B, 3.1.D, 3.1.F, and 3.1.G before selecting a pilot plant fabrication process.

The carbothermic reduction process, in which mixed oxides are reacted with carbon, at the present time seems to offer the best promise. Arc melting of metal and carbon, being used to process experimental fuel by WARD, is not amenable to high throughputs and yields a powder product which is difficult to sinter. The reaction of hydrides with carbon involves an additional processing stage (hydride formation) which adds to the cost of fabrication.

A low oxygen-bearing product may be difficult to obtain from the carbothermic reduction process, but whether or not this represents a difficulty depends on the evaluation of the effects of oxygen on fuel properties and behavior. An additional processing stage is required to produce Cr-modified carbide. This factor must be taken into account in selecting between a modified fuel or near-stoichiometric unmodified fuel.

Task 3.1.B - Fuel Properties Determination

The objective of this task is to characterize those physical and mechanical properties of mixed carbide fuel that are required to carry out fuel rod design and evaluate performance data. The effects of fabrication, composition, and irradiation variables on specific properties must be identified.

The bulk of this work will be out-of-pile on unirradiated materials. The effects of irradiation on properties must be determined on selected samples from irradiation rod tests.

The principal physical properties of mixed carbides on which data are essential are listed below in decreasing order (a through f) of importance:

- a. Thermal conductivity
- b. Self diffusion and fission product diffusion
- c. Melting points of matrix and second phases
- d. Specific heats
- e. Vapor pressure
- f. Thermal expansion

Fuel variables (stoichiometry, etc.) are unlikely to affect thermal conductivity to the extent where an optimum fuel is defined as one having the maximum conductivity, as is the case for mixed oxide fuels. However, rod design requires data on such effects as stoichiometry, second phase and porosity distribution, and sodium impregnation. The effects of fission product buildup (both solid and gaseous) during irradiation must be determined by post-irradiation studies and by instrumented irradiation tests.

Self-diffusion may affect the nuclear characteristics of a rod and fuel-bond-cladding compatibility. The process must be studied as part of the post-irradiation examination program and by classical diffusion experiments.

Melting of second phases (especially in modified fuels) may cause the maximum fuel temperatures to be limited to values much lower than the bulk melting temperature. These potential temperature limits must be determined to support appropriate irradiation experiments.

It is difficult to identify meaningful fuel mechanical properties that enter into the design and operation of an unrestrained, sodium-bonded fuel rod. The only significant stresses either are thermal in origin or result from the differential swelling properties of different temperature regions. The problem thus becomes one of irradiation creep. Specific irradiation tests are required to study this particular property.

A partially restrained rod concept may evolve during this development program. It may be possible to analyze fuel behavior in such a rod using creep properties. It is considered more likely however, that the behavior of the large volume of fission gas bubbles present at this stage will

dominate fuel-clad mechanical interactions. This latter type of behavior can only be adequately tested in appropriate irradiation experiments.

Task 3.1.C - Cladding Properties Determination (Unirradiated)

The objective of this task is to characterize 316 stainless steel and generate those data necessary for the establishment of 1) a material specification which will guarantee high quality reliable tubing for cladding, and 2) statistically significant engineering design properties representative of the tubing. An additional objective is to continue development of those alternate cladding materials discussed in the Task I report so as to permit a sound evaluation of their potential and limitations relative to those of 316 stainless steel. These materials include 347 stainless steel, Incoloy 800 and the vanadium-base alloys.

The function of the cladding is to contain the fuel and maintain the structural integrity of the fuel rod for the duration of the fuel rod lifetime and with a low probability of failure. The fuel rod lifetime dictated by cladding behavior must not be significantly different from that imposed by fuel behavior. The present state of technology and lack of required knowledge on stainless steels contribute to the dilemma of selecting a reasonable fuel rod lifetime. Failure to achieve the objectives of this task will heavily penalize the core design, by requiring the use of large uncertainty factors to account for unknowns in the reproducibility of batches of tubing and the missing engineering design values. To minimize these penalties and provide for the design of a highly reliable and economical fuel rod, the following experimental subtasks are required.

Subtask 3.1.C.a - Mechanical Properties

The objective of this subtask will be to determine through statistically planned testing programs the engineering design properties of 316 stainless steel. The properties of interest will be ultimate, yield and fatigue strengths, and ductility determined from accurate stress-strain curves, creep and creep-rupture and low-cycle fatigue. Test conditions will emphasize biaxial stress systems and the effects of

continuously increasing stress which simulates the stress buildup on cladding due to fission-gas release and swelling in fuel rods. The integration of stress and time gives a more realistic approach to the design basis than does the conservative constant stress test. Some correlation with uniaxial testing will be required to provide a less-costly means for comparison with the alternate materials. Tests will be performed in sodium to carefully assess the effects, if any, of the LMFBFR coolant on the properties. Temperatures of interest will be 900 - 1400°F with more emphasis on the higher temperatures where information is most lacking. Material for these tests will be representative of the products and results of the other subtasks in this task.

In addition to the detailed efforts on 316 stainless steel, carefully controlled screening tests will be carried out on the alloys considered as potential alternates for cladding to qualify or disqualify each material for detailed evaluation. Tensile and creep tests at two significant temperatures for 347 stainless steel, Incoloy 800, and, probably, two promising vanadium alloys will be sufficient.

Subtask 3.1.C.b - Metallurgical Stability

This subtask is designed to evaluate the various possible conditions of 316 stainless steel tubing and select that combination most complementary to the objectives of the main task. The evaluation will consider annealed versus cold-worked material, grain size, and structural variations in precipitates, inclusions, and existing phases. The effects of the more desirable conditions on mechanical properties will be determined under Subtask 3.1.C.a. The prime effort of this subtask will utilize metallographic and microscopic equipment to analyze variously prepared and treated samples.

Subtask 3.1.C.c - Fabrication and Process Control

This subtask will concentrate on the evaluation of fabrication and treatment processes for high quality tubing and the ultimate selection and specification of the processes necessary to achieve the goals of the other two subtasks. Material produced by the optimum procedures will be utilized in the final generation of engineering design properties.

The overall integration of these subtasks is complex, but is absolutely necessary to meet the objectives of the LMFBR program and the proper design of a fuel rod. In addition to these subtasks, sodium corrosion effects and design allowances must be determined for the cladding. General corrosion studies are described under a separate task on sub-assembly development. These data will be applicable to the cladding, provided temperatures as high as 1400°F are studied. Also, fuel-cladding compatibility effects on cladding performance are considered in a separate task of the fuel rod development.

Task 3.1.D - Fuel-Cladding Compatibility Studies

Chemical reaction between the fuel and cladding, with or without a contribution from the sodium bond, must not decrease the cladding life or seriously impair fuel behavior. The objective of this task is to show that candidate fuel-cladding combinations are satisfactory, according to the above definition, for up to 750°C and 25,000 hours. A totally inert fuel-bond-cladding combination is not an essential fuel rod requirement.

Two types of fuel-cladding reactions, requiring different testing techniques, must be studied.

a. Solid state reactions at points of physical contact:

Eutectic formation between free metal in hypostoichiometric fuel and iron-base cladding material exemplifies this type of reaction. Gas-filled capsules with intimate fuel-cladding couples will be used in these tests.

b. Mass transfer reactions in which a species is transferred through the sodium bond

Carbon transfer to the cladding is the most probable effect.

Materials will be tested in simulated fuel rod configurations.

Testing programs to study both types of reaction are required. The time and temperature dependence of reaction of various fuel-cladding combinations, and information on reaction mechanisms will be obtained from these tests. The effects of fission products in the bond may be simulated in out-of-pile tests.

All fuel rod irradiation tests will yield compatibility data with the emphasis on low exposure times and moderate temperatures. Temperature gradients and heat fluxes in the irradiation tests may modify mass transfer mechanisms.

Task 3.1.E - Development of Fuel Rod Fabrication Procedures

This task has the primary objective of developing techniques for the fabrication of LMFBR fuel rods. It will also provide suitable experimental fuel rods for irradiation testing under Tasks 3.1.F and 3.1.G and simulated rods for compatibility tests in Task 3.1.D.

The four major fabrication stages that require development effort are:

- a. Sodium filling
- b. Sodium bonding
- c. Sodium bond evaluation
- d. Rod sealing

Sodium filling may be done by a variety of methods, such as adding sodium pellets and subsequent melting, vacuum casting liquid sodium, or bottom filling with through-flow in a loop system. The principal factors are the long fuel-plus-blanket column involved and the desirability of eliminating fuel debris from the bond annulus.

Sodium bonding requires a thermal anneal at 500-600°C and also may require some mechanical forces to eliminate voids and gas bubbles. Various vibration modes or centripetal forces could be used for this purpose.

Eddy current inspection of sodium bonds currently is used for EBR-II driver fuel rods, experimental metal fuel rods and WARD experimental carbide fuel rods. The applicability of the method to very long rods has not been demonstrated. Alternate inspection methods, particularly those which can be applied to a molten bond, require evaluation.

Welding of end plugs should not raise any significant difficulty for rods clad with stainless steel. Because of the potential hazard of a bottom end failure followed by bond loss, very stringent inspection procedures will be required.

In all fabrication procedures, the economic and technical aspects of the alternates must be evaluated.

Task 3.1.F - Fast Flux Rod Irradiation

Fuel rods will be irradiated in fast flux environments at ratings, temperatures, and burnups as closely prototypical of the LMFBR as possible. Enrichment of the fuel and/or a high-flux facility will be required to obtain appropriate specific power ratings.

These objectives may be achieved in a three phase program. Phase I will test rods at moderate ratings (15-30 kw/ft) to burnups around 10^5 MWD/T. Rod diameters will be typical of the LMFBR design (about 0.3 inch diameter). In Phase II, ratings will be increased to the range 30-45 kw/ft. This can be achieved in EBR-II only by increasing the rod diameters to 0.4-0.5 inch. Phase III tests will be prototypical with respect to specific ratings, linear ratings, enrichment, temperatures and burnups. These conditions can only be attained in FFTF, or in the Demonstration Plants.

In all phases of the program, the tests will yield specific data on the following:

- a. General rod behavior
- b. Steady state failure mechanisms (not in Phase I)
- c. Fuel swelling characteristics
- d. Fission gas release behavior
- e. Formation of central void in fuel
- f. Sodium-bond performance (gas and solid contamination)
- g. Fuel-sodium-clad reactions
- h. Irradiation effects on the cladding

After testing, samples of fuel will be used to investigate the effects of burnup on properties such as thermal conductivity and melting point.

It may be hypothesized that as swelling increases with burnup, further swelling should become progressively easier to restrain. This argument leads to an alternate rod concept in which the fuel is restrained by

the cladding after it has swelled 15-25 percent. If this concept were valid, a significant increase in the smeared density of the fuel should result. Based on Phase I and II results, a subsidiary program of tests on potentially-restrained fuel rods should be designed and performed.

Task 3.1.G - Thermal Flux Rod Irradiation

Fast flux irradiation space will be overloaded until FFTF or Demonstration Plants become available for prototype rod and subassembly testing. Therefore, full use must be made of thermal flux test facilities to carry out those testing programs for which a fast flux environment, although desirable, is not absolutely essential.

Suitable programs for thermal flux testing would be the preliminary characterization of fuel material irradiation behavior, fuel-cladding compatibility, and the behavior of fission gases in the sodium bond. Studies of the effects of irradiation on properties could also be performed in a thermal flux.

It is difficult to design good prototype fuel rod tests without preliminary fuel swelling and gas release data. Suitable data could be generated from tests of small rods containing reference and alternate fuels. The latter would be evaluated in a comparative manner to select the most suitable specifications for prototype rod testing.

Fuel-cladding compatibility should not be affected by the power profile. Useful data in this area may be generated in thermal flux tests, provided that note is taken of the high surface burnup.

Surface burnup will also affect the release of fission gas and solid fission products into the sodium bond in tests designed to study their subsequent behavior.

Samples for the study of the effects of irradiation on properties can be extracted from other programs. Studies of fissile atom and fission product migration can also be conducted on this basis, provided appropriate temperature gradient corrections are applied.

Task 3.1.H - Transient Rod Testing

The objective of this task is to study the inherent safety characteristics of sodium-bonded carbide rods under fast reactor transient conditions. The specific tests will be designed to establish the limiting conditions, mechanisms, and consequences of various failure modes. This work will be conducted in two phases.

Phase I will consist of tests performed on individual fuel rods with various design concepts. The objectives of these tests will be to define and characterize the modes of fuel rod failure with respect to fuel rod parameters and reactor transient conditions. The variables that will be tested are:

- a. Fuel rod bond condition
- b. Fuel type - modified versus unmodified
- c. Fuel and rod dimensional parameters
- d. Vented versus non-vented fuel rods
- e. Fuel burnup (simulated and actual, up to 100,000 MWD/MT)
- f. Transient testing conditions (peak power and durations)

Phase II of transient tests will be conducted on clusters of fuel rods (approximately 3 rods per cluster) in static and flowing sodium. The objective of these tests will be to determine the consequences of a gross fuel rod failure, the propagation of the failure to adjacent rods, and the effects of rapid vaporization of the sodium coolant.

Task 3.1.I - Cladding Properties Determination (Irradiated)

The objective of this task is to determine the effects of neutron radiation exposure on the performance and characteristics of the cladding material. More specifically, the task aims to establish the statistically significant engineering design properties of 316 stainless steel representative of end-of-life conditions. This involves the generation of properties required for design as a function of neutron exposure, with additional consideration of the neutron energy spectrum. By achieving these objectives, the largest single source of uncertainty in safety factors for design can be eliminated.

The experimental studies under this task are long range in order to present results from exposures to 5×10^{23} nvt. Complete assessment of existing programs in EBR-II and anticipated results from FFTF and the Demonstration Plants will be factored into the effort to provide the necessary data. The subtasks would follow those of Task 3.1.C with the exception of Subtask 3.1.C.c, and would make use of material representative of the results of Subtasks 3.1.C.b and 3.1.C.c. The studies would be as follow:

Subtask 3.1.I.a - Mechanical Properties

These tests would parallel those of Subtask 3.1.C.a, but with considerably less effort devoted to the alternate materials. In-pile and post irradiation creep data will be compared to determine the significance of the latter in establishing design properties. The effects of void formation and growth on the end-of-life ductility will be determined, as will the effects of cladding stress on void growth.

Subtask 3.1.I.b - Metallurgical Stability

This subtask will be concerned with those items described under Subtask 3.1.C.b as they are affected by radiation. The interaction of cold work conditions and metallurgical phases with changes in threshold temperatures due to irradiation will be determined. Particular emphasis will be placed on the void formation effects on structure and on resultant changes in density and density-dependent properties.

Subtask 3.1.I.c - Corrosion in Sodium

This subtask will determine the extent to which neutron radiation affects the corrosion rates of 316 stainless steel. This could result in significant difference due to ionization and must be accounted for in the establishment of design allowances of material loss. Input for this subtask will depend on the results of investigations from FFTF and the Demonstration Plants, in addition to a few supplementary controlled experiments.

Coordination between Task 3.1.C and Task 3.1.I efforts will be maximized in order to utilize information from both in establishing design bases for the LMFBF.

3.1.4 Relation to LMFBR Program Plan

The LMFBR Program Plan, Volume 7, identifies four fuel rod concepts that currently appear to have an adequate potential for LMFBR use. They are:

- a. Gas-bonded oxide
- b. Gas-bonded carbide
- c. Sodium-bonded carbide
- d. Sodium-bonded metal

The sodium-bonded carbide proposed in the WARD 1000 MWe study theoretically has the capability of attaining the highest rating of the candidate concepts. This potential is recognized in the LMFBR Program Plan.

Type 300 stainless steels, type 316 in particular, appear as though they might be adequate for a first-generation 1000 MWe LMFBR, but their properties, rather than fuel performance, may limit core operation. Considerable emphasis therefore, is required on advanced alternates such as vanadium alloys. While these appear unlikely to be available for a 1980 commitment, sufficient effort on their development is required to make them available within a future 3-4 year period. The potential of alternates to type 300 stainless steels is not adequately acknowledged in the LMFBR Program Plan.

3.1.5. Major Test Facility Requirements

The test facilities to carry out the development programs proposed in this program are outlined below. Not all are new, but some expansion or modification of existing facilities may be required to properly implement the program.

Fuel Rod Fabrication Facility (Task 3.1.A)

The fuel rod fabrication facility will develop and utilize processes capable of being expanded to provide subassemblies for a Demonstration Plant, i.e., an output of about 200 rods/week. The additional expansion to LMFBR core fabrication would probably be attained by plant duplication.

Development input to this facility would be from a small-scale fabrication facility with the following characteristics:

- a. High purity atmosphere
- b. Ability to accommodate alternate fuel synthesis methods
- c. In-line fuel characterization
- d. Facilities for investigating alternate rod assembly and bonding techniques
- e. Rod Non-destructive testing facilities

Fuel Physical Metallurgy Facility (Tasks 3.1.B, 3.1.F and 3.1.G)

Fuel property studies (Task 3.1.B) require conventional physical metallurgy equipment housed in α -tight, inert-atmosphere gloveboxes. For similar work on irradiated fuels (Tasks 3.1.F and 3.1.G), the equipment will be contained in α - β - γ -shielded enclosures and operated remotely.

Cladding Mechanical Testing (Tasks 3.1.C and 3.1.I)

Conventional tensile, creep, fatigue, and tube burst testing equipment will be required for work under Task 3.1.C. In addition, similar equipment which has been modified to test materials under static and flowing sodium also will be required.

Work on irradiated cladding materials (Task 3.1.I) will need similar equipment in β - γ shielded cells.

Compatibility Test Facility (Task 3.1.D)

Compatibility capsules will be prepared in the fuel rod fabrication facility, and then tested and evaluated in the physical metallurgy facility. Additional special purpose facilities for tests under defect conditions will include small sodium loops and sodium vapor enclosures.

Fast Flux Test Facilities (Task 3.1.F)

EBR-II is available for fast flux studies but is limited to rod powers not exceeding 30-35 kw/ft because of the physical limit on rod size in

the standard 19-capsule subassembly. A new subassembly, containing 12 or 13 capsules, will be required to perform tests at 40-45 kw/ft.

High power tests in flowing sodium (bare rod tests) will require the loop facilities in FFTF or experimental subassemblies in a Demonstration Plant core.

Thermal Flux Test Facilities (Task 3.1.G)

Instrumented capsule irradiation facilities are available in several thermal flux test reactors. An adjustable facility (like the V-RAFT in GETR) will be required to maintain constant operating conditions in highly enriched fuels.

Transient Test Facilities (Task 3.1.H)

TREAT and SPERT currently provide the capability of testing with a wide range of transient power insertions. The principal shortcoming of these facilities is that the transient is superimposed over a zero power (uniform temperature) condition. It would be desirable to have the capability of superimposing a power insertion over normal operating power. This may be feasible by utilizing a rapid low-power to high-power location transfer in FFTF.

Post-Irradiation Examination Facilities (Tasks 3.1.F, 3.1.G and 3.1.H)

The post-irradiation examination facilities needed to evaluate rods must have the following essential features:

- a. α - β - γ shielding to contain 10^6 curies
- b. High purity inert (argon) atmosphere
- c. Special purpose equipment to:
 1. Measure rod and fuel dimensions
 2. Radiograph rods
 3. Extract fission gas samples
 4. Determine fuel density
 5. Perform chemical analyses on the fuel, cladding and bond
 6. Examine the fuel and cladding metallographically (optical and electron)

7. Identify minor phases in fuel and cladding (electron microprobe)
8. Prepare fuel and cladding samples for further experimentation
9. Re-encapsulate rods for additional steady state or transient irradiation testing

3.1.6 Program Schedule

The fuel rod development schedule is shown in Figure 3.1-2. A task-by-task description of the illustrated milestones is presented below.

Task 3.1.A - Fuel Fabrication Processes-Evaluation

1. Identify fuel fabrication process
2. Complete development of basic fabrication process
3. Complete development of scale-up pilot-plant level

Task 3.1.B - Fuel Property Determination

1. Complete Detailed Work Plans for the physical and mechanical properties test programs and initiate physical properties test program
2. Initiate mechanical properties test program
3. Complete physical properties test program
4. Complete mechanical properties test program

Task 3.1.C - Cladding Properties Determination (Unirradiated)

Subtask 3.1.C.a

1. Determine the availability of preliminary design data for 316 SS
2. Determine the availability of final design data for 316 SS
3. Complete evaluation of screening tests on alternate cladding materials
4. Determine the availability of design data for one alternate material (if screening tests show it better than 316 SS)

Subtask 3.1.C.b

5. Establish preliminary cladding condition requirements
6. Complete evaluation of cladding conditions -- set final requirements

Subtask 3.1.C.c

7. Establish preliminary fabrication and processing requirements for tubing
8. Complete evaluation of fabrication and processing procedures -- set final requirements for tubing

Task 3.1.D - Fuel-Cladding Compatibility Studies

1. Complete Detailed Work Plan for out-of-pile test program
2. Initiate out-of-pile test program
3. Complete out-of-pile test program
4. Complete detailed work plan for and initiate in-pile test program
5. Complete in-pile test program

Task 3.1.E - Development of Rod Fabrication Procedures

1. Identify procedures for experimental rod fabrication and inspection
2. Complete development of prototype rod fabrication

Task 3.1.F - Fast Flux Rod Irradiation

1. Initiate Phase I program in EBR-II
2. Complete Phase I program in EBR-II
3. Initiate Phase II program In EBR-II
4. Complete Phase II program in EBR-II
5. Initiate Phase III programs in FFTF and Demonstration Plant
6. Complete Phase III program on reference fuel rod concept

Task 3.1.G - Thermal Flux Rod Irradiation

1. Complete preliminary reference fuel behavior studies
2. Complete alternate fuel behavior studies
3. Complete program on irradiation effects in fuels

Task 3.1.H - Transient Rod Testing

1. Complete Detailed Work Plans for Phase I and II transient tests and initiate Phase I program
2. Complete Phase I and initiate Phase II transient test program
3. Complete Phase II transient test program

Task 3.1.I - Cladding Properties Determination (Irradiated)

Subtask 3.1.I.a

1. Establish preliminary design factors for radiation effects on mechanical property design data
2. Complete evaluation swelling studies of cladding -- set appropriate limits for cladding design properties
3. Establish final design factors to account for radiation effects

Subtask 3.1.I.b

4. Establish radiation effects on cladding condition requirements

Subtask 3.1.I.c

5. Set preliminary corrosion allowances
6. Set final sodium corrosion allowances, including effects of radiation, etc.

Figure 3.1-2. Fuel Rod Development Program Schedule

3.2 Core Structural Materials Development

3.2.1 Objective

To develop Core Structural Materials Technology to a level where the design and fabrication of units, which meet the functional requirements of the 1000 MWe LMFBR Reference Plant, can be accomplished without undue technical and economic risks.

3.2.2 Program Summary

The program is divided into seven major tasks aimed at developing information on materials for the following core structural applications:

- Fuel rod spacer grid
- Blanket assembly wrapper cans
- Core instrumentation
- Control and safety rod assemblies
- Core support and flow plates
- Core barrel
- Neutron shield reflector

3.2.3 Task Descriptions

Task 3.2.A - Fuel Assembly Structural Material Development

The objective of this task is to develop the required materials property data to verify that the structural components of the fuel assembly, specifically the spacer grid and the strongback, are properly designed for the environment in the Reference 1000 MWe LMFBR. The task is divided into two subtasks which are described below.

Task 3.2.A.a - Fuel Rod Spacer Grid Materials Development

The primary functions of a spacer grid are as follows:

- a. Hold the fuel rods in position and prevent vibration at temperatures up to 1200°F, at coolant flow rates of up to 30 feet/second
- b. Prevent thermal bowing

- c. Permit axial movement of fuel rods

To design and proof test the reference grid concept, the following materials data will be developed:

- a. The mass transfer and corrosion rates of candidate grid materials in flowing sodium will be determined in a series of sodium corrosion test loop runs simulating temperature and flow conditions in the reference design. Extrapolations of the test loop data will establish material loss rates for the core lifetime.
- b. The effects of sodium on the mechanical properties of candidate grid materials will be determined in specially designed test loops. Mechanical properties to be studied in this program will include stress relaxation and fatigue. Wherever feasible, the bulk of the sodium exposure experienced by the test pieces will be carried out in pre-exposure loops not fitted with mechanical testing equipment.
- c. Fabrication procedures could have a profound effect on the performance of the grids, particularly when brazing is used to produce grid joints. The effects of various brazing parameters will be evaluated to optimize the choice of the brazing alloy for this application. Although preliminary studies of this aspect could be conducted without the sodium environment, the final evaluation will require mechanical testing in sodium, as described above.
- d. Irradiation effects on mechanical properties of candidate grid materials, will be studied in specially designed subassembly tests in existing sodium cooled reactors. In-pile stress relaxation testing will be of particular importance in this part of the program.

Task 3.2.A.b - Fuel Assembly Strongback Materials Development

The active core fuel assembly design incorporates a strongback which replaces the wrapper can in other designs. This strongback compensates for the thermal stresses in the fuel assembly. To fully evaluate this concept, materials data will be developed as follows:

- a. Sodium corrosion resistance data and mass transfer characteristics of candidate strongback materials will be obtained by loop testing under conditions simulating the reactor core sodium environment. The design and operation of these loops will be similar to those currently being used to study other candidate primary system materials.
- b. Tests will be designed to evaluate the effects of irradiation on the mechanical and thermal properties of the strongback assembly materials. Redundant space in subassembly tests will be used for exposing the preliminary test pieces. Final evaluation will be accomplished by substitution of subassembly fuel rods by strongback rod assemblies.

Task 3.2.B - Blanket Assembly Can Materials Development

Blanket assembly designs incorporate a wrapper can that controls the coolant flow distribution and compensates for the thermal stresses developed in the subassemblies. The objective of this task will be to assure that the sodium and fast flux environment prevailing in the blanket areas will not adversely affect the performance of the wrapper can. Materials data requirements for the design of wrapper cans will to some extent parallel those for other primary system components. However, the flux and temperature in the blanket region will be lower. Interpolation of data will be facilitated by exposing materials test pieces in suitable regions of existing sodium cooled reactors.

Task 3.2.C - Core Instrumentation Materials Development

Reliable core instrumentation must be employed to monitor the reactor environment. Wherever possible, existing materials evaluation data will be used to assure reliable performance of the instrumentation. Materials used for instrumentation containment will be evaluated in other parts of the LMFBR Development Program, to minimize the number of materials requiring characterization in the reactor core environment.

Task 3.2.D - Control Rod Materials Development

The objective of this task is to identify and develop the materials data required to design suitable control and safety rod assemblies. Identification of materials data requirements will be made when the mechanical design has progressed to a more definitive stage. The development program for this task probably will be similar to that for Task 3.2.A.b. In addition, wear and galling tests will be required. Fabrication development requirements also will be identified as the mechanical design progresses, with the primary objective being to assure jointure reliability by mechanical testing.

Task 3.2.E - Core Support and Flow Plate Materials Development

Core support plates in the Enrico Fermi reactor have exhibited pitting damage resulting from misalignment induced cavitation. Fabrication and assembly procedures will be specified to minimize the problem, which could cause over-heating of blanket subassemblies and dispersion of stainless steel particles into the primary sodium system. An alternate approach to be evaluated is minimizing cavitation by hard-facing the core support plate. This approach would require sodium corrosion testing of suitable hard-facing alloys which could be accomplished in a conventional sodium corrosion test loop.

Fabrication techniques for flow guide plates will also be evaluated to insure that failure does not occur. The objectives of this task will be achieved by developing rigorous specifications for all flow plate fabrication.

Task 3.2.F - Core Barrel Materials Development

Because a failure in the core barrel could cause the collapse of the major core internals, materials and fabrication evaluations are aimed at assuring the reliability of this component. The effects of sodium environment and irradiation on the mechanical properties of welded AISI-304 must be understood before committing this component to service. Much of the sodium corrosion data obtained from the IHX materials program

will be applicable to the core barrel, avoiding a separate test program. However, irradiation data will have to be obtained from previous work on AISI-304 and supplemented with in-pile weldment evaluations. Detailed fabrication and materials specifications will be compiled.

Task 3.2.G - Neutron Shield/Reflector Materials Development

The reference selection of Inconel 600 as a reflector material requires that sodium corrosion and irradiation data be developed for this alloy. While the sodium corrosion testing of this alloy could be performed in a conventional test loop, an alternate approach would be cladding the Inconel 600 with a material having a relatively well known corrosion rate, such as AISI-304. Cladding techniques will be evaluated for the shielding materials which will be selected following further design development. Irradiation testing of shield reflector materials will be performed in specially designed subassemblies in the test facilities in existing reactors.

3.2.4 Relation to LMFBR Program Plan

Many of the topics discussed in the presented plan have been covered by various parts of the LMFBR Program Plan, as shown below:

Task 3.2.A

1. Task Area 5-1, "Materials Compatibility", in Volume 5
2. Task Area 7-12, "Structural Materials", in Volume 7
3. Task Area 6-2, "Technology of Fuel Element and Assembly Design", in Volume 6

Task 3.2.B

1. Task Area 5-1, "Materials Compatibility", in Volume 5
2. Task Area 7-12, "Structural Materials", in Volume 7
3. Task 6-3.1, "Blanket Design Technology", in Volume 6

Task 3.2.C

Tasks in Volume 4, "Instrumentation and Control", will be used to define materials requirements which will then be listed under Task Area 5-1 or Task Area 7-12.

1. Task Area 5-1, "Materials Compatibility", in Volume 5
2. Task Area 7-12, "Structural Materials", in Volume 7

Tasks 3.2.D and 3.2.G

1. Task Area 5-1, "Materials Compatibility", in Volume 5
2. Task 7-11.3, "Absorber Materials", in Volume 7
3. Task 6-3.2, "Control and Safety Assembly Design Technology", in Volume 6

Tasks 3.2.E and 3.2.F

1. Task Area 5-1, "Materials Compatibility", in Volume 5
2. Task Area 7-12, "Structural Materials", in Volume 7
3. Task 6-3.3, "Core Structure Design Technology"

3.2.5 Major Test Facility Requirements

Sodium test loops (v5) are needed for simulating reference plant thermal and flow conditions for sodium corrosion evaluation and determination of the effect of sodium on the mechanical properties of plant materials. Facilities of this nature are available at WARD and are described in Appendix A. Access to suitable positions in existing LMFBR plants is required for the specified irradiation testing programs.

3.2.6 Program Schedule

The core structural materials development schedule is shown in Figure 3.2-1. A task-by-task description of the illustrated milestones follows.

Task 3.2.A.a - Fuel Rod Spacer Grid Materials Development

1. Initiate grid materials sodium corrosion test loop.
2. Initiate grid materials mechanical properties in sodium loop.
3. Initiate evaluation of sodium corrosion and mechanical properties in sodium data.
4. Initiate irradiation testing of grid material samples to establish effect of irradiation on properties in sodium.
5. Initiate evaluation of irradiation testing.
6. Initiate grid performance testing under simulative plant conditions.
7. Complete grid materials evaluation program.

Task 3.2.A.b - Fuel Assembly Strongback Materials Development

1. Initiate strongback materials sodium corrosion loop and concept design verification study.
2. Begin irradiation testing of strongback materials.
3. Complete strongback materials program.

Task 3.2.B - Subassembly Can Materials Development

1. Issue topical report on performance of wrapper cans in sodium cooled reactors.
2. Complete 10,000 hour sodium corrosion tests (corrosion program at ARD started in 1967).
3. Initiate irradiation tests on simulative wrapper cans.
4. Complete wrapper can materials evaluation.

Task 3.2.C - Core Instrumentation Materials Development

1. Complete review of relevant materials data.
2. Identify required program.
3. Complete core instrumentation materials evaluation.

Task 3.2.D - Control Rod Materials Development

1. Complete literature review of control rod materials.
2. Initiate fabrication development and performance evaluations of wearing and galling of control rods.
3. Initiate irradiation testing of reference control rod materials.
4. Complete control rod materials evaluation.

Task 3.2.E - Core Support and Flow Plate Materials

1. Complete literature survey on core support and flow plate materials, including hard-facing materials.
2. Complete design and fabrication of loops to evaluate corrosion and pitting losses.
3. Complete evaluation of core support and flow plate materials.

Task 3.2.F - Core Barrel Materials Development

1. Complete literature survey on core barrel materials performance and fabrication.
2. Start testing of simulative weldments both in and out of pile.
3. Compile specifications for core barrel fabrication and issue final report on core barrel materials data.

Task 3.2.G - Neutron Shield/Reflector Materials Development

1. Complete literature survey on shield and reflector materials and define program plan.
2. Complete evaluation of shield and reflector materials.

Figure 3.2-1. Core Structural Materials Development Program Schedule

3.3 Steam Generator Materials Development

3.3.1 Objective

To develop the materials technology to the level where material performance under expected service and possible accident conditions and environments in the reference 1000 MWe LMFBR can be reasonably predicted.

3.3.2 Program Summary

The program is divided into nine major tasks. The spectrum covers corrosion and mechanical property determinations for structural materials both in water/steam and in sodium environments. The effect of fabrication processes on these properties is also included, and particular emphasis is placed on the performance of the tube-to-tube sheet weld.

3.3.3 Task Descriptions

Task 3.3.A - Sodium Corrosion of Steam Generator Materials

The objective is to obtain data on the sodium corrosion and erosion resistance of AISI Type 304 stainless steel and Incoloy 800 under representative service conditions. The operating conditions will include high sodium flow rates, temperatures up to 1050°F, temperature gradients throughout the system (hot and cold legs), and specific surface area to volume ratios. The results must be of sufficient reliability to allow reasonable extrapolation over the lifetime of the steam generator (30 years). Failure to obtain such data will mean that material limitations due to metal loss over extended periods of time will be uncertain and could lead to an inefficient and costly steam generator design to allow for this uncertainty.

Steam generator material specimens will be exposed in the cold leg of pumped sodium test loops which are as nearly as possible representative of the secondary system. In this manner, mass transfer data will be obtained under conditions typical of those expected in the steam generator. These data will be utilized in the development of an analytical corrosion model which will enable corrosion rates to be predicated under any specified conditions.

Task 3.3.B - Mechanical Properties of Steam Generator Materials in Sodium

The objective is to obtain data on the effect of sodium on the mechanical properties of AISI Type 304 stainless steel and Incoloy 800. As with the sodium corrosion data to be obtained in Task 3.3.A, mechanical property tests will be performed under sodium conditions representative of those expected in service. The main emphasis will be placed on creep, stress-rupture, stress-relaxation, and fatigue measurements, since these are areas in which little or no data are presently available under expected service conditions. Tensile properties (UTS, yield strength, elongation) will also be determined. Such data are essential so that design curves and limitations will be available for materials under expected service conditions.

Testing will be done in both stagnant (short term tests) and flowing (long term tests) sodium and in an inert atmosphere for comparison. In order to reproduce the surface conditions expected after prolonged exposure, specimens will be pre-exposed in flowing sodium under conditions determined by the mass transfer program. These specimens will then be tested in either flowing or stagnant sodium.

Since the effect of mass transfer is to alter the specimen surface, which, in turn, affects the mechanical properties, the volume to area ratio and specimen geometries are of obvious significance in any mechanical testing program. Wherever possible, therefore, specimens of the correct geometry will be used.

Task 3.3.C - Stress Corrosion Cracking of Steam Generator Materials

The objective is to determine the stress corrosion cracking resistance of Incoloy 800 in expected service and possible accident conditions. This involves screening tests in strong caustic solutions and other tests in water containing hydroxide, chloride, or oxygen ions. Both water and steam environments will be of interest.

Steam generator temperatures, pressures, and steam qualities will be reproduced in Inconel 600 autoclaves. The expected boiling temperature in the steam generator is 669°F; this will be used as the test temperature.

Internally pressurized tube specimens, stressed to 95 percent of the 0.2 percent proof yield stress at temperature, will be tested to failure in environments of varying hydride or chloride percentages as well as in typical operating chemistry environments. Testing will be done in dry steam, wet steam and solid water at or about the 669°F temperature.

Task 3.3.D - Corrosion Fatigue

The objective is to obtain data on the effect of water/steam environments on the fatigue properties of Incoloy 800. It is conceivable that such environments are detrimental to the fatigue resistance of the alloy, and it is essential to be aware of how much of a design limitation this might impose.

Steam generator tubing will be fatigue tested in push-pull type tests in inert gas at 1000°F and 669°F, in steam at 1000°F, and water/steam mixtures at 669°F. The effect of different water chemistries on the fatigue behavior of the steam generator tube materials will be studied in this manner.

Task 3.3.E - Mechanical Properties of the Tube-to-Tube-Sheet Weld

The objective is to determine the high temperature mechanical properties of the tube-to-tube sheet weld. The main interest centers on creep and fatigue properties, both in sodium and in water environments. The tube-to-tube sheet weld is an area of critical importance in the design of the steam generator, and failure to obtain such information will render the safety of the steam generator indeterminate.

The testing of tube-to-tube-sheet welds will be an integral part of the mechanical properties testing program, outlined in Task 3.3.B, and of the corrosion fatigue program, Task 3.3.D. Tube-to-tube-sheet welds will be evaluated under conditions similar to those indicated in those tasks.

Task 3.3.F - Sodium Corrosion of the Tube-to-Tube-Sheet Weld

The objective is to determine the corrosion of the tube-to-tube sheet weld in sodium under expected service conditions. As with the tube and the tube sheet materials, results must be of sufficient applicability

and reliability to allow predictions to be made of the expected metal loss due to corrosion over the life-time of the steam generator. Without such data, design limitations due to such loss will be subject to uncertainty.

Tube-tube-sheet weld specimens will be tested in the sodium corrosion program outlined in Task 3.3.A. The corrosion behavior will be determined for these welds and particular emphasis will be put on studying any effects of sodium on the weld itself and in the heat affected zones.

Task 3.3.G - Weld Defects in the Tube-to-Tube-Sheet Weld

The objective is to determine the effect of weld defects on the properties of the tube-to-tube sheet weld. In view of the critical nature of the weld in the overall design of the steam generator, it is important that information be obtained on the desired quality of the weld and possible detrimental effects arising from weld defects.

Weld defects in tube-to-tube-sheet welds will be identified and characterized in an extensive metallographic survey. The occurrence of these defects will be statistically determined and their effect on mechanical properties studied.

Task 3.3.H - Tube Wastage from a Sodium-Water Reaction

The objective is to obtain data on the effect on the integrity of the steam generator tubes on the appearance of a small water or steam leak in a tube or tube weld. This will include determination of the growth rate of the leak itself and also the effect of the leak on tubes in its immediate vicinity. The information will be relevant in estimating the amount and rate of extension of damage likely to ensue following the formation of a small leak, and thus indicating the extent of repairs that may be necessary.

High temperature-high pressure water or steam will be ejected from simulated tube and tube-to-tube-sheet weld leaks of known size and configuration into a reaction vessel containing high temperature sodium. The water/steam jet will impinge on a target specimen.

The wastage rates of the target material will be determined as a function of sodium temperature, jet temperature, flow size, and jet to target distance.

The increase in flow size with time will also be determined.

Task 3.3.I - Fabrication Processes

The objective is to determine the effect of fabrication processes on all the properties mentioned in the previous tasks. In view of the differences likely to occur during commercial production, it is important that these effects be identified at an early stage. This would reduce the possibility of failure at later stages.

Such variables as cold work introduced into tubing by drawing and bending, residual stresses in weld regions, and structural changes caused by welding may effect the performance of the materials. These factors will be included in the program outlined above and where detrimental effects are found steps will be taken to rectify them.

3.3.4 Relation to LMFBR Program Plan

The LMFBR Program Plan, Task Area 3-6, "Steam Generator," of Volume 3, recognizes the need for the generation of materials properties information similar to the plan proposed here. Among the parameters specifically identified, data will be required on the mechanical properties of materials as a function of time and environment. Knowledge of corrosion and erosion effects is also considered desirable. The tube-to-tube-sheet weld is identified as an area of greatest concern and the integrity of the weld should be thoroughly explored. In addition to these factors, it is also recognized that information is required on the effect of fabrication processes on materials properties, in particular the weld materials. The Program Plan further states that, even with maximum reliability, the possibility of a sodium-water reaction must be considered and efforts made to determine the effects of such a reaction with respect to tube wastage.

3.3.5 Major Test Facility Requirements

Sodium Corrosion (Task 3.3.A)

Two sodium loop systems with attendant control instrumentation and cover gas systems.

Mechanical Properties in Sodium (Task 3.3.B)

Tensile testing: 1 pre-exposure loop; high temperature, controlled atmosphere testing equipment.

Creep: stress-relaxation testing: 1 pre-exposure loop; high temperature, controlled atmosphere testing equipment; testing equipment incorporating a sodium loop.

Fatigue testing (both high cycle/low strain and low cycle/high strain): 1 pre-exposure loop; high temperature, controlled atmosphere testing equipment; testing equipment incorporating a sodium loop.

Stress Corrosion Cracking (Task 3.3.C)

Six autoclaves; testing equipment involving the use of internally pressurized tubular specimens.

Corrosion Fatigue (Task 3.3.D)

Two tension-compression fatigue machines; high temperature, high pressure test chambers.

Mechanical Properties of the Tube-to-Tube-Sheet Weld (Task 3.3.E)

One pre-exposure loop; mechanical testing equipment similar to that used in Tasks 3.3.B and 3.3.D.

Sodium Corrosion of the Tube-to-Tube-Sheet Weld (Task 3.3.F)

One sodium loop system with attendant control instrumentation and cover gas system.

Weld Defects on the Tube-to-Tube-Sheet Weld (Task 3.3.G)

Same as specified for tasks 3.3.E and 3.3.F

Tube Wastage from a Sodium-Water Reaction (Task 3.3.H)

Sodium vessel, with attendant control instrumentation and cover gas system; pressurized steam-water loop.

Fabrication Processes (Task 3.3.I)

One sodium corrosion loop with attendant control instrumentation and cover gas system; mechanical testing equipment as for Task 3.3.B.

3.3.6 Program Schedule

The steam generator materials development schedule is shown in Figure 3.3-1. A task-by-task description of the illustrated milestones is presented below:

Tasks 3.3.A and 3.3.F - Sodium Corrosion of Steam Generator Materials

1. Beginning of MTL program
2. Completion of 10,000 HR run at max conditions
3. Final topical on MTL program
4. Data on Tube-to-Tube-Sheet Welds
5. Final corrosion model

Tasks 3.3.B and 3.3.E - Mechanical Properties of Steam Generator Materials in Sodium

1. Initiate program
2. Begin pre-exposure
3. Begin out-of-sodium tests
4. Begin sodium tests
5. Make available initial tensile data
6. Make available initial creep data
7. Make available initial fatigue data
8. Complete program

Task 3.3.C - Stress Corrosion Cracking of Steam Generator Materials

1. Initiate program
2. Complete facility installation
3. Make available screening test data
4. Prepare preliminary data for Incoloy 800

5. Install loop
6. Complete autoclave work
7. Complete loop work

Task 3.3.D - Corrosion Fatigue

1. Initiate program
2. Commence testing
3. Prepare data for operating chemical environment
4. Prepare data for abnormal chloride and hydroxide environments
5. Present final report

Task 3.3.G - Weld Defects in the Tube-to-Tube-Sheet Weld

1. Initiate program
2. Commence statistical survey of weld defects
3. Complete task

Task 3.3.H - Tube Wastage from a Sodium-Water Reaction

1. Initiate task
2. Commence testing
3. Prepare data on reference tubing
4. Prepare data on tube-to-tube-sheet welds
5. Complete task

Task 3.3.I - Fabrication Processes

1. Initiate task
2. Review state-of-the-art
3. Perform interim evaluation
4. Complete final fabrication report

Figure 3.3-1. Steam Generator Materials Development Program Schedule

3.4 Intermediate Heat Exchanger Materials Development

3.4.1 Objective

To develop the materials technology to the level where material performance under expected service and possible accident conditions and environments in the reference 1000 MWe LMFBR can be reasonably anticipated.

3.4.2 Program Summary

The program is divided into three major work areas, mainly covering materials performance in a flowing sodium environment. Consideration also is given to the effect of fission products, in the primary coolant, on the structural materials.

3.4.3 Task Descriptions

Task 3.4.A - Sodium Corrosion of IHX Materials

The objective is to obtain data on the sodium corrosion and erosion resistance of AISI Type 304 stainless steel under representative service conditions. The material will be exposed to high sodium flow rates and temperatures, and temperature gradients (in the form of hot and cold legs in a pumped sodium loop). Efforts will be made to recreate the specific surface-area-to-volume ratios encountered in practice. The results must be of sufficient reliability to allow reasonable predictions of the metal loss, due to corrosion, likely to occur over the anticipated lifetime of the intermediate heat exchanger. Lack of such data could lead to inefficient and costly design in order to allow for the uncertainty in predictions based on other information.

IHX materials will be exposed in the cold leg of pumped loops, making up the primary sodium circuit and in the hot leg of loops representing the secondary sodium circuit. In this manner, mass transfer data will be obtained under conditions typical of those expected in the IHX. These data will be utilized in the development of analytical corrosion model which will enable corrosion rates to be predicted under any specified conditions.

Task 3.4.B - Mechanical Properties of IHX Materials in Sodium

The objective is to obtain data on the effect of sodium on the mechanical properties of AISI Type 304 stainless steel. As with the sodium corrosion data to be obtained in Task 3.4.A, mechanical property tests will be performed under sodium conditions representative of those expected in service. The main emphasis will be placed on creep, stress-rupture, stress-relaxation, and fatigue measurements, since these are areas in which little or no data is presently available under the relevant sodium conditions. Tensile properties (UTS, yield strength, elongation) also will be determined before and after sodium exposure. Such data are essential in order that design curves and limitation will be available for material under expected service conditions.

As with the steam generator (Task 3.3.B), the IHX materials will be tested in stagnant and flowing sodium and in an inert gas for comparison. The effects of long term exposure will be estimated by using pre-exposed specimens and testing them in stagnant or flowing sodium. Again, great importance will be attached to specimen geometry, and appropriate geometries will be used whenever possible.

Task 3.4.C - Effect of Fission Products on Materials Properties

The objective is to obtain some measure of the effect that fission products in the primary sodium coolant system might have on the structural and mechanical properties of intermediate heat exchanger materials. This would involve testing as performed under Tasks 3.4.A and 3.4.B, but using sodium contaminated with fission products.

Separate mass transfer and mechanical testing facilities would be required. It is not intended that the complete mass transfer and mechanical property programs be repeated using fission product contaminated sodium. Selected experiments would be run which would best indicate any effects that may arise.

3.4.4 Relation to LMFBF Program Plan

The LMFBF Program Plan, in Task Area 3-12, "Heat Exchangers" in Volume 3, recognizes the need for the generation of materials properties information similar to the plan proposed here. Among the parameters specifically identified, data will be required on the mechanical properties of materials as a function of time and environment. Knowledge of corrosion effects also is considered desirable. The possible effects of fission products on the structural and mechanical properties of materials is not envisaged in the Heat Exchanger Task Area of the Program Plan, but Volume 5, Task Area 5-3, "Fission Products in Sodium Systems", gives consideration to this problem.

3.4.5 Major Test Facility Requirements

Sodium Corrosion (Task 3.4.A)

A sodium loop system with attendant control instrumentation and cover gas system as was described for Program 3.3.

Mechanical Properties in Sodium (Task 3.4.B)

Tensile testing: One pre-exposure loop; high temperature, controlled atmosphere testing equipment.

Creep testing: One pre-exposure loop; high temperature, controlled atmosphere testing equipment; testing equipment incorporating a sodium loop.

Fatigue testing: Two pre-exposure loops (both high cycle); high temperature, controlled atmosphere testing equipment; testing equipment incorporating a sodium loop (low cycle/high strain).

Effect of Fission Products on Materials Properties (Task 3.4.C)

Testing equipment as described under Task 3.4.B but in which all the sodium systems contain contaminated sodium.

All of these facilities exist at WARD and are described in the Appendix.

3.4.6 Program Schedule

The intermediate heat exchanger materials development program schedule is shown in Figure 3.4-1. A task-by-task description of the illustrated milestones is presented below.

Task 3.4.A - Sodium Corrosion of IHX Materials

1. Beginning of MTL Program.
2. Completion of 10,000 hour run at maximum conditions.
3. Final topical on MTL Program.
4. Data on tube-to-tubesheet welds.
5. Final corrosion model.

Task 3.4.B - Mechanical Properties of IHX Materials in Sodium

1. Initiate program.
2. Begin pre-exposure.
3. Begin out-of-sodium tests.
4. Begin in-sodium tests.
5. Make available initial tensile data.
6. Make available initial creep data.
7. Make available initial fatigue data.
8. Complete program.

Task 3.4.C - Effect of Fission Products on Materials Properties

1. Initiate task.
2. Install equipment.
3. Prepare preliminary corrosion data.
4. Prepare mechanical property data.
5. Complete task.

Figure 3.4-1. Intermediate Heat Exchanger Materials Development Program Schedule

3.5 Large Sodium Pump Materials Development

3.5.1 Objective

To develop sodium pump technology to a level where the design and fabrication of units which meet the functional requirements of the reference 1000 MWe LMFBR plant can be accomplished without undue technical and economic risks.

3.5.2 Program Summary

Four areas have been identified in which materials information is needed.

- Mechanical property data are required for structural materials in sodium before design criteria can be identified.
- Suitable bearing materials need to be identified and proof tested.
- Fabrication techniques for large bearings, impellers, shafts, and casings need to be resolved.
- Corrosion-erosion data are required for pump impeller materials in high velocity sodium.

These areas are covered by four tasks outlined below which are intended to produce the required information.

3.5.3 Task Descriptions

Task 3.5.A - Mechanical Property Data Determination

Design codes used for the fabrication of components for sodium service must be based on mechanical properties data obtained in sodium, because some of these mechanical properties may be expected to change in this environment.

For the operating temperature range, 700-800°F, the properties of interest are principally tensile properties and fatigue properties. At these temperatures, creep and stress rupture data are difficult to obtain but can be estimated from higher temperature data.

Tensile data should include measurement of yield stress, ductility, and ultimate stress both for short and long pre-exposure times.

Fatigue data are required up to 10^7 - 10^8 cycles, again with long pre-exposure times and without pre-exposure.

The program outlined probably will be run as part of the general mechanical properties in sodium program, including Tasks 3.3.B and 3.4.B for the steam generator and IHX respectively.

Task 3.5.B - Bearing Material Selection

Several important properties govern the selection of a bearing material for in-sodium applications. These are:

- a. Sodium corrosion and mass transfer
- b. Wear behavior in sodium
- c. Thermal properties (conductivity, expansion)

Candidate materials, selected on the basis of available property data, should be wear-tested under simulated start-up and low load conditions in liquid sodium at operating temperatures.

Mass transfer tests will be carried out in pumped loops capable of attaining velocities as seen by the bearings. Corrosion rates will be determined as a function of temperature, velocity and oxygen level in the sodium.

Sliding wear tests will be run in flowing sodium, and the best materials combinations selected. The selected combination then will be proof tested in a rotating type test facility in which start up and low load conditions can be reproduced. A survey of physical properties will be carried out and programs identified in this area.

Task 3.5.C - Fabrication

The pump design for the 1000 MWe plant represents a substantial increase in size over existing sodium pumps. Because of this, fabrication techniques currently in use may not be adequate for the manufacture of key components such as shafts, bearings, and impellers.

A program, therefore, is needed to determine the suitability of current techniques for the production of large components, and, where necessary, to identify and perfect new fabrication processes.

Initially, a state-of-the-art review will be made, and the applicability of current techniques to the fabrication of these large size components determined. Where necessary, techniques will be modified or new ones perfected. After proving on an experimental scale, full size components will be fabricated and tested.

Task 3.5.D - Corrosion-Erosion Data for Pump Impeller Materials

The relative sodium velocity at the impeller tip is of the order of 80-100 fps. At these high velocities, it is possible that the metal loss rate at the impeller tip, due to combined corrosion and erosion, might be enhanced. A program, therefore, is required to determine this effect under pump operating conditions.

A pumped loop will be constructed capable of achieving velocities up to 100 fps at 1200°F through the test section. Candidate impeller and diffuser materials will be tested in the as-cast, annealed, and sensitized conditions over a range of velocity temperatures and oxygen levels. Weight losses will be determined as a function of time.

3.5.4 Relation to LMFBR Program Plan

All tasks are covered in one or more sections of the LMFBR Program Plan. Obviously, a substantial amount of overlap in materials requirements for various components occurs in the Program Plan, and some of the tasks are also applicable to other components.

The Program Plan emphasizes the need for in-sodium mechanical property data over the entire temperature range experienced during plant operation in Volumes 3, 5, and 7. The properties listed above would be determined as part of a comprehensive mechanical property program. Similarly, bearing development and fabrication procedures are covered in the Program Plan.

3.5.5 Major Test Facility Requirements

Mechanical Properties Test Facility (Task 3.5.A)

This facility should be capable of:

- a. Accepting tensile and fatigue specimens
- b. Operating under flowing sodium conditions over a large temperature range
- c. Running with a controlled oxygen content
- d. Maintaining a suitable ΔT

In addition, pre-exposure facilities are required. These types of facilities are available at WARD and are described in the Appendix.

Wear of Bearing Materials Test Facility (Task 3.5.B)

The wear test facility should be a rotating device in which start-up and low-load conditions can be reproduced. The sodium environment should be as for Task 3.5.A.

Mass Transfer in High Velocity Sodium Requirements (Task 3.5.D)

- a. Pumped loop capable of achieving up to 100 fps through the test section
- b. Temperatures in excess of 800°F.
- c. Oxygen level control.

3.5.6 Program Schedule

The large sodium pump materials development program schedule is shown in Figure 3.1-2. A task-by-task description of the illustrated milestones is presented below.

Task 3.5.A - Mechanical Property Data Determination

- 1. Initiate program.
- 2. Begin pre-exposure.
- 3. Begin out-of-sodium tests.
- 4. Begin sodium tests.
- 5. Make available initial tensile data.
- 6. Make available initial creep data.
- 7. Make available initial fatigue data.
- 8. Complete program.

Task 3.5.B - Bearing Material Selection

1. Initiate program.
2. Complete literature review.
3. Initiate testing.
4. Make available comparative corrosion data.
5. Complete the available wear-data program.

Task 3.5.C - Fabrication

1. Initiate program.
2. Complete state-of-the-art review.
3. Begin development work.
4. Scale up new techniques.
5. Complete program.

Task 3.5.D - Corrosion-Erosion Data for Pump Impeller Materials

1. Initiate program.
2. Initiate testing.
3. Prepare preliminary data for reference conditions.
4. Complete program.

Figure 3.5-1. Large Sodium Pump Materials Development Program Schedule

3.6 Check Valve Materials

3.6.1 Objective

To develop sodium check valve technology to a level where the design and fabrication of units which meet the functional requirements of the reference 1000 MWe LMFBF plant can be accomplished without undue technical and economic risks.

3.6.2 Program Summary

Materials information is required in the following areas:

- Materials need to be identified for the structural components, hard facings, and spring materials.
- In-sodium mechanical property data are required for structural materials before design criteria can be identified.
- Valve seat materials require testing under sliding wear conditions in sodium.
- Fabrication techniques need to be determined.

These areas are covered by the tasks outlined below.

3.6.3 Task Descriptions

Task 3.6.A - Materials Identification

Materials have not yet been identified for the valve components. The structural material will be governed by the material used for the primary loop construction to minimize incompatibility problems. Hard facing selections, however, should take into account the data generated in Task 3.6.C.

A complete review of potential materials will be made and potential materials selected. Final selection should be made on the basis of the mechanical property program and the identification of the hard facing materials program (Tasks 3.6.B and 3.6.C below).

Task 3.6.B - Mechanical Property Data

Design codes used for the fabrication of components for sodium service must be based on mechanical property data obtained in sodium since some of these mechanical properties may be expected to change in this environment. For the operating temperature range, 700-800°F, the properties of interest are principally tensile and fatigue properties. At these temperatures, creep and stress rupture data are difficult to obtain but may be estimated from higher temperature data.

Tensile data will include measurement of yield stress, ductility, and ultimate stress, both for short and long pre-exposure times.

Fatigue data are required up to 10^7 - 10^8 cycles, again, with and without pre-exposure to sodium.

In-sodium mechanical property, data will be generated as part of the general in-sodium mechanical property program already outlined in Tasks 3.3.B, 3.4.B and 3.5.A. The requirements are essentially the same as those for the pump.

Task 3.6.C - Wear of Hard Facing Materials

All candidate materials should be tested under conditions simulating operating bearing loads, total wear path length, and operating environment. Suitable materials should be selected accordingly.

Wear testing will be carried out on potential hard facing materials in flowing sodium. Bearing loads, wear path lengths, and wear frequencies will be reproduced as closely as possible. Wear will be measured as a function of time, temperature, and oxygen level.

Task 3.6.D - Fabrication

Due to the substantial increase in size over existing check valves, fabrication techniques currently in use may not be adequate for the manufacture of valve components. This applies, particularly, to the application of the hard facing to the valve seat. A program is necessary, therefore, to determine the suitability of current techniques for the production of

large components, and, where necessary, to identify and perfect new fabrication processes.

Initially, a state-of-the-art review will be made, and the applicability of current techniques to the fabrication of these large size components determined. Where necessary, techniques will be modified or new ones perfected. After proving on an experimental scale, full-size components will be fabricated and tested.

3.6.4 Relation to LMFBR Program Plan

The LMFBR Program Plan embraces all aspects of materials development and application, and the tasks listed above are all covered in one or more sections of the Plan. Structural material development programs for various components obviously overlap, and the mechanical properties in sodium work outlined is applicable to other components, for example the pumps. The Program Plan emphasized the need for in-sodium mechanical property data over the entire temperature range experienced during plant operation in Volumes 3, 5, and 7, and the properties listed in Task 3.6.B would be determined as part of a comprehensive mechanical property program.

Fabrication and hard facing development are also covered in the Plan.

3.6.5 Major Test Facility Requirements

Mechanical Property Test Facility

This facility should be capable of:

1. Accepting tensile and fatigue specimens
2. Operating under flowing sodium conditions over a large temperature range
3. Running with a controlled oxygen content
4. Maintaining a suitable ΔT

In addition, suitable pre-exposure facilities are required. Facilities of this type exist at WARD and are described in the Appendix.

Wear Test Facility

A facility is needed in which the expected sliding conditions at the valve seat can be reproduced. Requirements are:

1. Friction coefficients and wear rates can be measured.
2. The test can be carried out in flowing sodium at operating temperatures.
3. Long term tests can be run.

3.6.6 Program Schedule

The check valve materials development schedule is shown in Figure 3.6-1. A task-by-task description of the illustrated milestones is presented below.

Task 3.6.A - Materials Identification

1. Initiate program.
2. Survey available materials.
3. Begin preliminary selection.
4. Complete final selection.

Task 3.6.B - Mechanical Property Data

1. Initiate program.
2. Begin pre-exposure.
3. Begin out-of-sodium tests.
4. Begin sodium tests.
5. Make available initial tensile data.
6. Make available initial creep data.
7. Make available initial fatigue data.
8. Complete program.

Task 3.6.C - Wear of Hard Facing Materials

1. Initiate program.
2. Commence testing.
3. Prepare preliminary data for reference conditions.
4. Complete program.

Task 3.6.D - Fabrication

1. Initiate program.
2. Complete the state-of-the-art review.
3. Begin development work.
4. Scale-up new techniques.
5. Complete program.

Figure 3.6-1. Check Valve Materials Program Schedule

3.7 Piping "Catastrophic"

3.7.1 Objectives

To study piping "catastrophic" corrosion and to define conditions under which it occurs. To make recommendations for its avoidance on the reactor, using this information.

3.7.2 Program Summary

The program mainly consists of determining the effect of several variables (temperature, atmosphere, composition) upon the attack by sodium-sodium compound mixtures on steel. Suggestions as to how to avoid this attack on the reactor after a leak occurs will be made.

3.7.3 Task Descriptions

Task 3.7.A - Experimental "Catastrophic" Corrosion Studies

Present reactor concept places much sodium containment in a nitrogen atmosphere. Tests will be made on a series of steel specimens, with a variety of nitrogen-based atmospheres. A range of sodium-sodium compound mixtures will be used at specific temperatures. Rates of attack will be determined from weight changes and thickness measurements. It is anticipated that the experiments will be "bench-top" in scale; the specimens will be fabricated into some form of holder, such as a shallow tray.

At a later stage, the other materials, such as those deriving from lagging, will be introduced and their effects determined.

Task 3.7.B - Development of Recommendations for Avoidance of "Catastrophic" Corrosion

The data obtained in Task 3.7.A will be used to prepare recommendations for avoiding "catastrophic" corrosion on the reactor. Likely time interval after a leak springs which would be available for leak detection and remedial action will be specified.

3.7.4 Relation to LMFBR Program Plan

The LMFBR Program Plan, Volume 5, calls for investigation of the kinetics of propagation of small sodium leaks progressing to massive deterioration of containment walls from the outside. No particular approach to the problem is suggested. The present program is seen as complying with the general requirement, but it omits specific leakage studies, such as defining the helium leakage rate above which sodium penetration of a wall occurs.

3.7.5 Major Test Facility Requirements

None

3.7.6 Program Schedule

The time table for this development program is presented in Figure 3.7-1. Major milestone definitions, as indicated in scheduling figures, are presented below:

Task 3.7.A - Experimental "Catastrophic" Corrosion Studies

1. Initiate studies of piping "catastrophic" corrosion.
2. Facilities for the tests assembled.
3. Tests of sodium-sodium compound mixtures completed.
4. Additional tests with other materials (lagging) completed.

Task 3.7.B - Development of Recommendations for Avoidance of "Catastrophic" Corrosion

1. Begin evaluation of corrosion data.
2. Recommendations completed.

TASK	TITLE	CALENDAR YEAR						
		1969	1970	1971	1972	1973	1974	1975
3.7.A	EXPERIMENTAL "CATASTROPHIC" CORROSION STUDIES	1	2	3	4			
3.7.B	DEVELOPMENT OF RECOMMENDATIONS FOR AVOIDANCE OF "CATASTROPHIC" CORROSION			1	2			

Figure 3.7-1. Piping "Catastrophic" Corrosion Studies Program Schedule

4. REACTOR DEVELOPMENT

4.1 Studies of Temperature Coefficient of (U, Pu)C Fuel

4.1.1 Objective

To determine the temperature coefficients of reactivity of the reactor. For safe operation of an LMFBR a non-positive, and preferably negative, total temperature coefficient is thought to be required. The relative contributions from thermal expansion effects of the structural materials and Na coolant in the vicinity of 523°C and from the Doppler effect are required. With regard to the latter, the sign and magnitude of the contribution from the fissile Pu isotopes in (U, Pu)C fuel rods containing up to 20 percent Pu are required.

4.1.2 Program Summary

The program is divided into five major tasks dealing with development of better ways of predicting and measuring the magnitude of the neutron spectrum in the resonance region, the study of the effect of the U/Pu ratio on the magnitude of the Doppler coefficient, its resolution from the effects of thermal expansion of fuel, and various proof tests needed to verify the accuracy of experimental methods.

These tasks are described in detail in the following section.

4.1.3 Task Description

Task 4.1.A - Effect of U, Pu/Na Ratio on Resonance Spectrum

The objective of this Task is to determine the effects of the U, Pu/Na ratio on the neutron spectrum in the energy range of importance for the Doppler effect. The ability to predict the magnitude and sign of the Doppler effect requires, among other things, the capability for calculating and measuring the relative magnitude of the neutron spectrum in the resonance region (0.1 to 10 keV). Physically, the magnitude depends on the volume ratio of U, Pu relative to the structural materials and Na. The predictions of the spectral shape are hampered by the lack of resolution in the measurements of the resonance structure of nuclei in that

region and the low state of development of spectrometers covering these energies. In this Task, therefore, better methods for measuring spectra in the Doppler region will be developed. The effects of the U, Pu volume fraction, and of heterogeneity, on the neutron spectrum in the resonance energy region will be measured. A range of compositions typical of LMFBRs will be covered in these spectral investigations. The measured spectra will be compared with their prediction obtained by means of computer codes such as the MC² and RABBLE.

Task 4.1.B - Effect of U/Pu Ratio on Magnitude of Doppler Coefficient

The objective of this Task is to determine the Doppler coefficient for many combinations of Pu/(U + Pu) ratios and for many isotopic compositions of Pu. Considerable controversy exists concerning the magnitude and sign of the Pu-239 contribution to the Doppler effect. Even less is known about the effect of Pu-241, and as both of these combine with the effect of U-238 (which is well understood) and of Pu-240 (still uncertain) to produce the net Doppler effect, the need exists for additional effort in this area. The resonance parameters of the Pu isotopes will be determined up to higher energies and with better resolution, possibly taking the advantage of high resolution offered by the underground detonation experiments.

The magnitude and sign of the Pu contribution to the Doppler effect will be measured in critical experiments for Pu/(U + Pu) ratios of interest (up to 30%) and for Pu isotopic compositions, varying from weapons-type material (low Pu-240, etc.), to that from long irradiations (high Pu-240, etc.). Methods of calculation will be checked against the results of these experiments.

Task 4.1.C - Spatial Distribution of Doppler Effect

The design of the 1000 MWe LMFBR has multizoned cores, each with a different U/Pu fraction. The predictions of the Doppler effect for such geometries would gain much in accuracy if they were supported by the experimental verification of the effect. The determination of the spatial dependence of the Doppler coefficient will be made in this Task.

Critical experiments will be performed in which whole core zones, rather than single rods, are heated and the magnitude of the Doppler effect is measured. The different core zones will have U/Pu ratios representative of the values to be used in the LMFBR. Measurements of this type will combine the determination of neutron importance in the various regions with the effect of spatial variation of the Doppler coefficient and, thus, will enable a better prediction of the effect to be made.

Task 4.1.D - Resolution of Temperature Effects

The objective of this Task is to determine separately the components of the temperature coefficient of reactivity. The temperature coefficient of a core fueled with (U, Pu)C and cooled with Na has components due to: (1) thermal expansion of the core structure, (2) change in density of Na, (3) thermal expansion of fuel, and (4) the Doppler effect in the fuel. Resolution of the total effect into the components to obtain their relative magnitudes has proven difficult. Experiments will be performed in which the fuel is thermally insulated from the core structure to obtain resolution between thermal expansion of the fuel support and the effect of heating the fuel itself. This will facilitate greatly the determination of the magnitude of the Doppler effect by difference. It is important to achieve resolution because the Doppler effect acts promptly with effectively zero time lag, hence, its importance for preventing prompt nuclear transients. For an actual reactor, thermal expansion of fuel rods in the axial direction changes the core height, whereas radial expansion affects primarily the fuel/Na ratio. The radial core expansion is only a secondary effect. The thermal expansion of the core is approximated only in critical experiments using split-table machines, with fuel and Na loaded in the form of wafers into holding drawers. In this instance, the thermal expansion does not change the fuel/Na ratio. The SEFOR experiments will be studied for applicability to this subject. If they are lacking, zoned critical experiments will be performed to isolate the components of the temperature coefficients of reactivity.

Operation of the Demonstration Plant will provide verification of the predicted temperature coefficients.

Task 4.1.E - Measurement of Doppler Effect in Fast Reactors

The objective of this Task is to measure the Doppler coefficient in several fast reactors. Measurement of the Doppler effect in the planned sequence of reactors of smaller size, such as the SEFOR, FFTF and the Demonstration Plant will resolve the remaining doubts concerning the magnitude and the importance of the effect in controlling the 1000 MWe LMFBR. The values predicted on the basis of calculations, and cross-checked with the results of critical experiments, will be verified by the measurements in these small reactors. The departures from these conditions in the 1000 MWe LMFBR will relate mostly to higher fuel operating temperatures arising from the anticipated higher fuel ratings, and the effects of multizone enrichment dispositions.

4.1.4 Relationship to the LMFBR Program

This research and development program complements the Reactor Core Engineering program described in Volume 6, Task Area 6-1 of the LMFBR Program Plan, with more emphasis given to resolving the various temperature effects, particularly the contribution to the Doppler effect from Pu-239 and higher isotopes. The following Task Areas of the LMFBR Plan also are pertinent:

- 9-1 Acquisition of Nuclear Data
- 9-4 Analysis of Neutronic Characteristics of Reactors and Fuel Cycles

4.1.5 Major Facilities and Equipment Required

Computing Facility (Tasks 4.1.A and 4.1.B)

CDC-6600 digital computer and peripheral equipment, or equivalent.

Critical Facilities (Tasks 4.1.A through 4.1.D)

Critical facility of ZPR or ZPPR type is required, plus equipment for measurement of neutron spectra and Doppler coefficients. The critical facilities will be modified for zoned experiments.

Fast Reactor Facilities (Task 4.1.E)

Fast reactors loaded with ceramic fuel are needed for measurement of the Doppler coefficient of the fuel at realistic conditions of power, burnup, etc.

4.1.6 Program Schedule

The time schedule for the examination of problems related to the Doppler effect in the core for inclusion in the design of the 1000 MWe LMFBR is presented in Figure 4.1-1. The major milestones are defined below.

Task 4.1.A - Effect of U, Pu/Na Ratio on Resonance Spectrum

1. Evaluate methods of measuring resonance spectra.
2. Initiate lattice tests with lattice variations.
3. Initiate analytical evaluation of lattice tests.
4. Complete experimental and analytical spectral work.

Task 4.1.B - Effect of U/Pu Ratio on Magnitude of Doppler Coefficient

1. Initiate lattice tests with variable U/Pu ratios.
2. Complete experimental lattice tests.
3. Complete evaluation of lattice tests and comparison of calculational methods.

Task 4.1.C - Spatial Distribution of Doppler Effect

1. Initiate critical experiments with heated reactor zones.
2. Complete the critical experiments.
3. Complete evaluation of the zonal experiments.

Task 4.1.D - Resolution of Temperature Effects

1. Design wet critical experiment facility.
2. Initiate wet critical experiments.
3. Complete wet critical experiments for resolution of temperature effects.
4. Complete analytical evaluation of experiments.
5. Complete overall verification in Demonstration Plant.

Task 4.1.E - Measurement of Doppler Effect in Fast Reactors

1. Initiate Doppler effect measurements in SEFOR.
2. Initiate Doppler effect measurement in FFTF.
3. Complete Doppler effect measurements in FFTF.
4. Initiate temperature effect measurements in Demonstration Plant.
5. Complete temperature effect measurements in Demonstration Plant.

Figure 4.1-1. Studies of Temperature Coefficient of (U,Pu)C Fuel Program Schedule

4.2 Development of Na Fuel Bond

4.2.1 Objective

To develop a successful Na bond for LMFBR fuel. This requires thorough knowledge of the behavior of a thin film of Na subject to gas passage and high temperature gradients. No formation of stationary pockets of gas along the length of the fuel and in the space between it and the cladding can be permitted because this could result in boiling of sodium or melting of fuel.

4.2.2 Program Summary

The program is divided into six tasks which deal with measurement of the needed thermodynamic and transport properties of Na contaminated by fission products and at temperatures up to fuel melting, as well the behavior of the bond in transients and the determination of failure thresholds. These Tasks, listed below, are described in more detail in the next section.

- Study of Na Pressure Surges in the Bond
- Measurement of Thermodynamic Properties of Na
- Preparation of Tables of Thermodynamic Properties of Na
- Measurement of the Effect of Na Impurities on Transport Properties
- Determination of Equilibrium Constants of Na Isomers
- Measurement of the Effect of Impurities in Na on Thermal Conductivity

4.2.3 Task Description

Task 4.2.A - Study of Na Pressure Surges in the Bond

The objective of this study is to develop an analytical model which describes the location and behavior of sodium in the fuel rods. In steady state operation of fuel rods, loaded with pelletized ceramic fuel and bonded with Na to the cladding, the Na penetrates into the gaps between the pellets and may reach the region of high temperature at the center of the rod. At high power ratings, the temperature at the center of the fuel rod might be sufficiently high to maintain a region of Na vapor.

When the fuel undergoes a nuclear transient or overpower condition, this vapor region could expand and subject the Na region between the pellet and the cladding to a pressure surge. The mechanism and speed with which this pressure surge is propagated, the extent of expulsion of Na bond, and the probability of rupture of cladding are all unknowns at the present. These factors must be studied before the successful operation of this concept can be assured. An adequate analytical model will be formulated to describe the location and the thermodynamic condition of the Na bond in a transient.

Task 4.2.B - Measurement of Thermodynamic Properties of Na

The objective of this Task is to determine the thermodynamic and transport properties of sodium at temperatures of up to 2800°C. Knowledge of the thermodynamic and transport properties of Na at temperatures up to and including the melting of fuel enhances the success in predicting the behavior of the Na bond. These properties will be measured up to 2400°C (melting temperature of UC). With these data the behavior of the Na bond at overpower conditions in which center melting occurs in the fuel rods can be calculated, and studies of the equation of state of the molten core in gross accident conditions can be made. At the present time, the data on thermodynamic properties of Na extend up to 1900°C only. The studies proposed will extend the thermodynamic properties of Na up to, and preferably beyond, the critical point.

Task 4.2.C - Preparation of Tables of Thermodynamic Properties of Na

The objective of this Task is to prepare tabulations of the thermodynamic properties of sodium. Simultaneously with the extension of measurements of the thermodynamic and transport properties of Na, complete tables of these properties will be prepared as printed tabulations and as computer subroutines. Since the studies of the thermodynamic condition of the constituents of the core in the event of gross meltdown are hampered by lack of data of properties of Na at high temperatures and pressures, prediction of these properties based on general compressibility factors and

the like, will be included in the tables. These will serve, in the interim period, before the results of measurements become available.

Task 4.2.D - Measurement of the Effect of Na Impurities on Transport Properties

The objective of this Task is to measure the transport properties of sodium contaminated by fission gases, fuel debris, and other impurities. There are indications that the impurities in Na can change the transport properties of Na significantly. Since the operation of a Na bond involves the exposure of Na to contamination by fission products, the prediction of the behavior of the Na bond may be grossly in error if their effect is not taken into account. When subjected to the action of an expanding fission gas, Na vapor bubble, or a pressure surge, the effects of impurities in changing surface tension, contact angle, and viscosity of Na become particularly important in considering the behavior of thin films of Na, which might be present in the bond.

In the experiments for measuring these effects, the impurities introduced into the Na will represent the chemical composition of fission products plus powdered or particulate (U, Pu)C, such as might be introduced into the Na bond by the crumbling of the fuel pellets.

Task 4.2.E - Determination of Equilibrium Constants of Na Isomers

The objective of this Task is to measure the equilibrium constants of the sodium isomers. The thermodynamic properties of Na vapor are affected by its isomeric form, which changes from monomer to dimer or trimer depending on temperature and pressure. The proportion of each isomer in the mix is measured by the respective equilibrium constants. Present knowledge of these constants and their dependence on pressure is very incomplete, making the theoretical predictions of the thermodynamic properties of Na vapor difficult. Consequently, these constants will be determined.

Task 4.2.F - Measurement of the Effect of Impurities in Na on Thermal Conductivity

The objective of this Task is to measure the thermal conductivity of the sodium bond as affected by changes in wetting and in material conductivity induced by impurities in the sodium. There are indications that impurities in the Na bond will affect the heat transfer through the bond in two ways: (1) by changing the thermal conductivity of the bond itself, and (2) by affecting the capacity of Na to wet the surfaces of the fuel pellet and of the cladding. The latter effect will change the thermal conductance of the interface between the two materials and make them dependent on the condition of the heat transfer surface. Experimental data on these phenomena are scanty, and the proposed experiments are needed to supply the required additional information.

4.2.4 Relation to LMFBF Program Plan

This program is similar in scope to parts of Task Area 6-2, "Technology of Fuel Element and Assembly Design", of the LMFBF Program Plan, Volume 6, Core Design. It places more emphasis on the performance of Na bonding, effects of impregnation of (U, Pu)C by Na, and the extension of measurements of thermodynamic and transport properties of Na.

This program is related also to Task 10-2.8, "Materials Interaction Phenomena for Accident Analysis", and Task 10-2.9, "Physical and Mechanical Properties for Accident Analysis", in Volume 10, Safety, of the LMFBF Program Plan.

4.2.5 Major Facilities and Equipment Required

Computing Facility (Tasks 4.2.A and 4.2.C)

CDC-6600 digital computer and peripheral equipment, or equivalent.

Sodium Bond Test Facility (Tasks 4.2.D and 4.2.F)

A small, general purpose laboratory with bench size equipment specifically designed (1) to study bubble motion in annuli with various fluids, and (2) to study sodium impregnation of carbide fuels.

Gap Conductance Test Facility (Tasks 4.2.D and 4.2.F)

A small sodium loop equipped with specially designed heated elements would serve to verify sodium gap conductance models during the preliminary development stage. A loop similar to the proposed Westinghouse TF-2 (see Appendix), with a flow capacity of 4 gpm, could be used.

Thermodynamic Properties Test Facility (Tasks 4.2.B and 4.2.E)

A test facility is needed for measuring thermodynamic properties of Na up to its critical point (2573°K, 350 atm) and preferably up to the melting temperature of heavy metal ceramics.

4.2.6 Program Schedule

The schedule for the development of the Na fuel bond is shown in Figure 4.2-1. It is arranged to assure accumulation of all information needed for designing Na bonded fuel of a 1000 MWe LMFBR. Major milestones of the program, as shown in Figure 4.2-I, are defined below.

Task 4.2.A - Study of Na Pressure Surges in the Bond

1. Design and start construction of experimental rig.
2. Start tests.
3. Complete tests and formulate an analytical model.

Task 4.2.B - Measurement of Thermodynamic Properties of Na

1. Start experimental program.
2. Conclude first phase of experimental program.
3. Issue evaluated results.

Task 4.2.C - Preparation of Tables of Thermodynamic Properties of Na

1. Issue tables based on available results with extrapolation.
2. Incorporate results of tests and remove extrapolation.
3. Issue complete computer subroutines.

Task 4.2.D - Effect of Na Impurities on Transport Properties

1. Design experimental rig.
2. Complete experiments.

3. Complete analysis.

Task 4.2.E - Determination of Equilibrium Constants of Na Isomers

1. Design experimental rig.
2. Start experiments.
3. Conclude experiments and analysis.

Task 4.2.F - Effect of Impurities in Na on Thermal Conductivity

1. Design experimental rig.
2. Commence experiments.
3. Complete experiments.
4. Complete analysis.

Figure 4.2-1. Development of Na Fuel Bond Program Schedule

4.3 Determination of Hydraulic Effects on Fuel Cladding

4.3.1 Objective

To determine the behavior of the cladding material as effected by flowing sodium. Successful operation of the reference LMFBR requires the assurance of cladding integrity with a sufficient margin of safety to withstand over-power conditions. The selection of the cladding thickness, therefore, must be evaluated with respect to its ability to withstand the hydraulic environment as well as the neutron and corrosive environments which are evaluated in Section 3.1 of this report.

4.3.2 Program Summary

The program is divided into three tasks dealing with the investigation of the ability of selected cladding material to withstand the degrading effects of mechanical and hydraulic abrasion. Appropriate tests in the FFTF and Demonstration Plant are included. These tasks are described in the next section.

4.3.3 Task Description

Task 4.3.A - Determination of Cladding Erosion by Cavitation

The objective of this Task is to minimize the effects of corrosion caused by cavitation by measuring the effects for several design configurations and choosing satisfactory ones. One unknown in the cladding design is how flow-related phenomena contribute to the cladding erosion. With high Na flow velocities, the trailing edges of the fuel-supporting grids may induce cavitation. The magnitude of the effect, its relation to assembly and fuel rod geometries, and its dependence on Na temperature and pressure will be investigated in detail. Sodium tests loops, of size sufficient to accommodate fuel assemblies, will be used, and the full range of velocity, temperature, heat flux, and pressure will be explored thoroughly.

Task 4.3.B - Selection of Cladding Thickness in Vented Design

The objective of this Task is to determine the minimum cladding thickness required for the fuel rods. An advantage of the vented cladding design is

that, through the release of fission gas, the required cladding thickness is reduced. Fuel rod thickness with rigidity sufficient to withstand deformation at high operating temperatures and coolant flow velocities will be determined. In addition, the minimum cladding thickness which feasibly can be manufactured and still provide failure-free operating performance will be determined. On the basis of these determinations, the cladding thickness will be able to withstand suitable overpower conditions. Tests will be performed to verify that the selected cladding thickness is satisfactory.

Task 4.3.C - Evaluation of the Effect of Grid Fretting

The objective of this Task is to determine the effects of grid fretting on the cladding. In fuel designs which are based on support of fuel rods by grid structure, the effect of grid fretting on the integrity of cladding might pose some problems. In regard to the performance of the grids, the unknowns relate to the retention of spring properties of the holding fingers at high irradiations, the effect of the flow induced vibration, and the coefficient of friction between the grids and cladding in the presence of Na. Tests in sodium loops will be performed to provide preliminary information. Ultimately, tests will be performed in FFTF and the Demonstration Plant environments.

4.3.4 Relationship to LMFBR Program

In the LMFBR Program Plan, the development of the vented cladding concept for carbide fuel bonded with sodium is related to Sub-Tasks under Task Area 6-2, Fuel Element and Assembly Design. Similar aims and objectives are given in Task Area 7-2, Design Data for Near-Term Fuel Elements, Sub-Task 7-7.2.1, Cladding Development and Improvement for Long-Term Missions, and Task Area 7-9, Cladding Technology Development of the LMFBR Fuels and Materials Program Plan. Equally important is Task 10-2.4, Fuel Failure and Melting of the Safety Program Plan.

4.3.5 Facilities and Equipment Required

Computing Facility (Task 4.3.A)

CDC-6600 digital computer and peripheral equipment, or equivalent are required.

Cladding Integrity Test Facility (Tasks 4.3.A and 4.3.B)

A facility capable of testing cladding under anticipated static and transient loads in a simulated environment is needed. The Westinghouse FREF-1 facility (see Appendix) will have this capability.

Flow Test Facility (Tasks 4.3.B and 4.3.C)

A small sodium loop, similar to the proposed Westinghouse TF-2 (see Appendix), with a flow capacity of 4 gpm could be used for testing promising venting mechanisms. A loop similar to GPL-1 or GPL-2 (see Appendix) will be needed for extensive fretting studies.

Mechanical Test Facility (Task 4.3.C)

A small, general purpose mechanical laboratory is needed, equipped with a universal testing machine, mechanical shaker, photoelastic equipment, etc.

4.3.6 Program Schedule

The schedule for the determination of hydraulic effects on fuel cladding is shown in Figure 4.3-1. Its arrangement assures the acquisition of needed data for cladding design on a proper schedule for both the Demonstration Plant and the 1000 MWe LMFBR. Major milestones of the program, as shown in Figure 4.3-1, are defined below.

Task 4.3.A - Determination of Cladding Erosion by Cavitation

1. Design and construct test rig.
2. Start tests.
3. Conclude tests.
4. Conclude analysis of results.

Task 4.3.B - Selection of Cladding Thickness for Vented Design

1. Insert non-vented fuel in the Demonstration Plant.
2. Conclude irradiation.
3. Conclude analysis.

Task 4.3.C - Evaluation of the Effect of Grid Fretting

1. Insert specimens in sodium loops.
2. Remove specimens from sodium loops for analysis.
3. Remove Demonstration Plant assemblies for examination.
4. Start examination.
5. Conclude examination.

Figure 4.3-1. Determination of Hydraulic Effects on Fuel Cladding Program Schedule

4.4 Studies of the Behavior of the Na Coolant

4.4.1 Objective

To gain an understanding of sodium boiling, its initiation, and its effects on reactor behavior. Safe operation of the LMFBR requires that the coolant be prevented from boiling, but should boiling occur, the boiling voids should not cause gross core melt down. The development of core design which will prevent void formation or limit it to safe proportions is necessary. To accomplish this objective requires an analytical description of the void formation supported by experimental verification.

4.4.2 Program Summary

The program is divided into eight tasks dealing with the investigation of the following topics:

1. The evaluation and formulation of hot channel factors
2. The effect of surface condition on the onset of boiling
3. An evaluation of canned fuel assemblies for preventing void propagation
4. Abnormal coolant temperatures caused by inlet or local leakage in a canned fuel assembly
5. Possibility of gross void limitation through modular design of the reactor
6. Time and space dependencies in void formation
7. Effect of gas entrainment
8. Destructive tests

These tasks are described in detail in the next section.

4.4.3 Task Description

Task 4.4.A - Formulation of Hot Channel Factors

The objective of this Task is to formulate hot channel factors appropriate to the reactor design. The core design hot channel factors will account for the problems specific to the design of the Westinghouse LMFBR. In particular, such location dependent effects will be included as the existence of crossflow in fuel assemblies without outside cans, the variation in the inlet flow distribution, power peaking, U/Pu zonal variation, etc. Due to the effect of decreased thermal conductivity of (U, Pu)C with higher Pu content, the latter effect especially can shift the position of the hot spot in the fuel away from the position of maximum flux to the position of maximum Pu concentration. The various factors mentioned above will be combined into a hot channel factor as the physical situation demands, with provision for statistical variation where applicable. Techniques also will be developed in order to calculate core failure probabilities under varying reactor conditions.

Task 4.4.B - Determination of the Effect of Surface Condition on Initiation of Na Boiling

The objective of this Task is to determine the effects of parameters such as surface condition and pressure on the initiation of sodium boiling. At the present state of the art, the data to explain sodium boiling characteristics are limited. These studies indicate that, where pure sodium is heated in a clean system, considerable superheat can be imparted to Na, and that when that unstable equilibrium is disturbed, an explosive flashing to vapor occurs. There are reasons to believe that in more representative conditions, with Na containing impurities, flowing in turbulent conditions, and contacting rough surfaces, the tendency to superheating will be suppressed. If such were the case, a much safer design would result. Therefore, experiments will be conducted in representative geometries, flow conditions and heat transfer surface preparation, to determine as closely as possible the mechanism of onset of Na boiling and the degree of superheat. The majority of sodium boiling studies have

been conducted at subatmospheric pressures. System pressure in LMFBR reactors ranges from 15 to 50 psia. Since the superheat required to initiate boiling is pressure-dependent, the proposed tests will be conducted at anticipated reactor pressure and subcooling conditions. Based on acoustic or other techniques, the methods of detecting the onset of boiling in the opaque liquid will be developed at the same time.

Task 4.4.C - Evaluation of Canned and Uncanned Fuel Assemblies with Regard to Effects of Flow Blockage and Void Propagation

The objective of this Task is to compare canned and uncanned fuel assemblies relative to inlet flow blockage and void propagation. Analytical studies and experiments will be conducted to compare the canned and uncanned fuel assemblies in regard to effects of flow blockage and sodium void propagation. Current thought is that the uncanned fuel assembly prevents boiling due to inlet flow blockage. In fact, preliminary calculations indicate that flow redistribution into a blocked fuel assembly occurs rapidly in an uncanned assembly lattice. The important question concerning propagation of boiling voids and local pressure buildup, however, is unresolved for both types of fuel assemblies. The choice between the two concepts requires better understanding of the process of void formation in Na and its propagation throughout the core. If the uncanned fuel assembly geometry does have a tendency to propagate sodium voids, the probability of void propagation will be assessed versus the probability of flow blockage and possible critical flow in a canned fuel assembly.

Task 4.4.D - Study of Time Sequence in Voiding Transients

The objective of this Task is to develop a model which describes the kinetic behavior of the reactor during voiding. Predictions of the kinetic response of the reactor in voiding transients requires knowledge of the time sequence in which the individual components of the core heat, melt, vaporize and are moved by the pressure surge. The time constants associated with these processes are dependent on heat transfer properties and the heat capacities of the constituents of the core. Cladding

failure through melting may have a different effect on void propagation than cladding rupture following a pressure surge in the bond, and the severity of accident may be different in each case. A model will be formulated to enable better understanding of the process and the ability to predict the sequence of events. Testing of the model is discussed in Task 4.4.H.

The magnitude of the maximum reactivity insertion which can be handled by the reactor automatically through temperature coefficients will be determined, and design criteria set for maximum allowable void.

Task 4.4.E - Study of Space Dependence of Void Effect

The objective of this study is to measure the sodium void reactivity as a function of position in the reactor. Voiding of the core has a position-dependent effect on reactivity, with the central regions having a positive and the outside regions a negative effect. A determination of the spatial dependence of the reactivity due to voiding will be carried out experimentally, preferably using a fuel geometry similar to that in a power reactor. Dry critical experiments, using the split table and fuel drawer technique, do not represent adequately the neutron-streaming present in a fuel rod bundle geometry. The use of zoned, critical experiments will be studied to save time and resources. If these do not suffice, then more representative core mockups, possibly using a wet zero power assembly, will be used with voids introduced by voiding complete canned assemblies. For either the zoned critical experiments or the wet critical experiments, the spatial dependence of reactivity due to voiding will be measured.

Task 4.4.F - Study of Modular Geometry for Void Limitation

The objective of this Task is to measure the sodium void coefficient, the control rod worths for voided and non-voided modules, and the kinetic coupling of modules. The modular design of the reactor is based on the ability of a modular geometry to reduce the magnitude of the positive effect of a Na coolant void. In addition, it is hoped that the void once formed will be localized in one module and that the presence of a

low-power blanket separating the modules will prevent the propagation of the void from module to module. The first of these objectives will be tested in a critical assembly, such as the ZPPR. Experiments utilizing at least two modules, and preferably three of four, will be performed and the magnitude of the void coefficient will be measured. Kinetic coupling between a voided and a non-voided module will be determined, and the control rod worths in the voided and non-voided (but adjacent) modules will be measured. The number of modules will depend on prior analytical work, as well as on the ability of the critical assembly to handle the large lateral dimensions of a modular core.

Task 4.4.G - Determination of the Effect of Entrained Gas and Local Boiling

The objective of this Task is to assess the possibility of gas entrainment in the coolant; to devise ways of preventing such entrainment; and to test these solutions. Because of the positive effect of voiding the center of the core, a danger exists that the presence of a large entrained gas bubble in the Na coolant stream might lead to a reactivity transient. The possibilities of gas entrainment in the Na coolant will be assessed, and ways will be devised to prevent such entrainment. Devices will be developed and incorporated in the design which will assure that the presence of a large gas bubble would not lead to damaging the core and subsequent meltdown.

If voiding is a problem, experiments on entrainment of gas bubbles in Na will be carried out, and degassing equipment will be developed for insertion into the coolant stream. Such equipment will be subject to experiments to assess effectiveness. The effects of sodium boiling and production of large volumes of Na vapor are, of course, similar. Although vapor bubbles in a subcooled Na stream will collapse partially and produce less effect than insoluble gas, they must be avoided.

Task 4.4.H - Destructive Tests for Determination of Energy Releases

The final assessment of the safety of any developed LMFBR design rests on the execution of suitable tests aiming at destroying the core in a

transient. When the adequate Pu inventories become available, tests of the SPERT type are needed and should be made. Properly instrumented wet tests with Na should be carried out, and the critical assembly pulsed by blowing gas bubbles through the core or by ejecting control rods. The energy releases in each of the separate tests should be determined, and a careful postmortem analysis carried out. Tests with control rod insertion counteracting gas void should be carried out to determine the respective time constants and the energy releases.

4.4.4 Relationship to LMFBR Program

This program is related to Task Area 6-2 of the Core Design Volume of the LMFBR Program Plan (Volume 6). Task Area 10-5, Facilities, of the LMFBR Safety Program Plan covers the major facilities needed to carry out most of this work. These, together with the bench type and larger loops of the type existing at Westinghouse, should meet the needs of the program. Of particular importance is the transient reactor facility planned under Task 10-5.3 of the LMFBR Program. The high-urgency Task 10-2.3, Sodium Boiling of the LMFBR Safety Program Plan is also important.

4.4.5 Major Facilities and Equipment Required

Computing Facility (Tasks 4.4.A, 4.4.B, and 4.4.H)

Digital computer, CDC-6600 and peripheral equipment, or equivalent.

Heat Transfer Facility (Task 4.4.B)

A sodium test loop, similar to Westinghouse TF-2 facility (see Appendix) is needed for study of effect of surface conditions at the onset of Na coolant boiling.

Voiding Sequence Test Facility (Tasks 4.4.D and 4.4.G)

A large water loop for hydraulic and vibrational testing of scale models of fuel assemblies and core modules is required. Tests of velocity profiles and crossflow experiments with gas blown in to simulate sodium vapor voids can be performed in a test loop similar to the Westinghouse Loop H-1.

This loop has a flow capacity of 8000 gpm (see Appendix). Instrumentation for recording time sequences in voiding transients should be provided at a sufficient number of points in the models of the core in order to yield information on the space-time distribution of voiding phenomena under conditions of high velocity coolant flow or coastdown flow. Syphon-breaking devices also should be tested.

Critical Facility (Tasks 4.4.E and 4.4.G)

A critical facility with the capacity to accommodate large cores, such as the ZPPR, is needed for tests of modular geometry and of the effect of blanket thickness on reducing the void effects and their propagation throughout the reactor. Modification may be required for zoned critical experiments.

Energy Release Test Facility (Tasks 4.4.C and 4.4.H)

An experimental facility is needed for measurement of energy releases in nuclear transients in a wet, fast assembly. A SPERT-type facility, properly equipped to handle sodium, is required. Although such a facility would form a major item in the costs of carrying out the LMFBR Plan, it is believed that such a facility is of the greatest importance.

4.4.6 Program Schedule

The schedule for the studies of the behavior of the Na coolant is shown in Figure 4.4-1. The studies refer to the determination of the behavior of the Na coolant in reactor environment which are needed for the development of the 1000 MWe LMFBR. The major milestones of the program, as shown in Figure 4.4-1, are defined below.

Task 4.4.A - Formulation of Hot Channel Factors

1. Formulate (U, Pu)C conductivity dependent hot channel factors for replacement cores of Demonstration Plant.
2. Verify postulated hot channel factors versus operating data from Demonstration Plant.
3. Re-examine and re-apply the new formulations to the 1000 MWe LMFBR.

Task 4.4.B - Determination of the Effect of Surface Condition on the Onset of Na Boiling

1. Design and construct experimental rig.
2. Commence tests.
3. Analyze results.

Task 4.4.C - Evaluation of Canned and Uncanned Fuel Assemblies with Regard to Effects of Flow Blockage and Void Propagation

1. Design and construct test rig for out-of-reactor tests.
2. Initiate out-of-reactor tests.
3. Complete out-of-reactor tests.
4. Design and construct the in-reactor test assemblies.
5. Commence tests in STF.
6. Conclude tests and analysis.

Task 4.4.D - Study of Time Sequence in Voiding Transients

1. Design and build test rig.
2. Commence out-of-reactor loop tests.
3. Conclude tests in water loop.
4. Commence tests in Na loop.
5. Conclude tests in Na loop.
6. Conclude analysis of results.

Task 4.4.E - Study of Space Dependence of Void Effect

1. Initiate split-table dry critical experiments.
2. Conclude experiments.
3. Conclude analysis, first phase.
4. Start wet critical experiments.
5. Conclude wet critical experiments.
6. Conclude analysis.

Task 4.4.F - Study of Modular Geometry for Void Limitation

1. Prepare ZPPR for modular experiments.
2. Commence static experiments.

3. Conclude kinetic experiments.

4. Conclude analysis.

Task 4.4.G - Determination of the Effect of Entrained Gas and Local Boiling

1. Develop degassing equipment for coolant stream.

2. Design and build test rig for insertion into a flow loop.

3. Conclude loop tests of degassing equipment.

4. Design equipment for transient tests in STF.

5. Commence transient voiding tests.

6. Conclude tests and analysis.

Task 4.4.H - Destructive Tests for Determination of Energy Releases

1. Design test rig.

2. Commence destructive tests.

3. Conclude tests and analysis.

Figure 4.4-1. Studies of the Behavior of Na Coolant Program Schedule

4.5 Development of Fuel Assembly

4.5.1 Objective

To develop a fuel assembly which will adequately perform under the anticipated conditions of temperature, mechanical arrangement, heat transfer, and vibration. A fuel assembly will be developed which will operate with minimum vibration in coolant velocities up to 30 ft/sec and coolant temperatures up to 1300°F (700°C). The assembly design will allow operation with power densities up to 490 kw/liter, and with transverse temperature gradients of 180°F (100°C). The assembly will minimize lateral thermal distortion and will accommodate an axial support or holddown mechanism.

4.5.2 Program Summary

The five tasks constituting this program investigate:

- Thermal stability
- Measurements of thermal bowing
- Determination of film coefficient correlation and hydraulic resistance
- Areas subject to cavitation
- Vibration characteristics of the assembly

Temperature-compensated strongbacks are dealt with in these tasks. These tasks are described in the next section.

4.5.3 Task Description

Task 4.5.A - Determination of Thermal Bowing and Thermal Stability of an Assembly

The objective of this Task is to determine the amount of thermal bowing and its effect on the thermal performance and stability of the fuel assemblies. The amount of thermal bowing a fuel assembly experiences and the parameters which affect thermal bowing will be determined first in an out-of-pile heat transfer - mechanical facility. To ascertain the effects of the complex thermal gradients to which a fuel assembly is subjected in a reactor environment, properly instrumented fuel assemblies

will be placed in the Demonstration Plant and operated at full power ratings.

Task 4.5.B - Determination of Film Coefficient Correlations

The objective of this Task is to measure the heat transfer coefficients for the fuel rods in the exact geometry to be used in reactor operation. Heat transfer in a liquid metal system is very sensitive to boundary conditions. Large tolerances, bowed fuel rods and/or displaced fuel rods can lower the heat transfer coefficient significantly. Out-of-pile thermofluids tests will be conducted to determine the effect of non-uniform fuel rod geometry on heat transfer. Tests also will be made to ensure that hot cladding spots do not exist in the vicinity of grids or other type of spacers. The heat transfer tests will be conducted out-of-pile under steady state and transient flow and power conditions.

Task 4.5.C - Evaluation of "Strongback" Performance in Large Thermal Gradients

The objective of this Task is to test the temperature-compensated strongback-eggcrate fuel support structure. The Westinghouse design of the fuel assembly for the core incorporates four strongback rods. They consist of concentric tubes of Mo and 304 SS; in operation they take advantage of the differential thermal expansion of these materials to compensate for the thermal gradients imposed on the assembly. The performance of these temperature compensators in realistic conditions has not been tested. In operation, the strongbacks and the eggcrate fuel-support structure are intended to keep the fuel rods undisturbed, regardless of the position of the assembly in the core or the transient thermal conditions. The structure should retain its rigidity and fuel rod aligning properties in 20% overpower conditions. This will be confirmed by appropriate tests, and the design of the assembly structure either will be confirmed or modified. The tests will be carried out in an operating reactor, such as FFTF or the Demonstration Plant, in order to obtain the proper combination of temperature, temperature gradient, stress, and neutron fluence.

Task 4.5.D - Measurement of Vibration and Hydraulic Resistance

The objective of this Task is to measure the vibration characteristics and hydraulic resistance of the fuel assembly. An important characteristic of the assembly design and one which is difficult to calculate accurately is the hydraulic resistance offered by the complex grid structure of the assembly to the coolant flow. For this reason, the design of the assembly will be tested by placing partial and full assemblies in water loops and measuring the hydraulic resistance under steady-state and transient-flow conditions. The vibration characteristics of the assembly will be determined as a function of flow velocity, spacer support distance, temperature and system pressure.

Task 4.5.E - Study of Velocity Profiles, Coolant Mixing and Cavitation

The objective of this Task is to measure the velocity distribution in the fuel assembly. Because of the tightness of the flow passages in the fuel assembly, the measurements of velocity profiles of coolant throughout the assembly is difficult. For this reason, oversized models of the assembly in a water loop will be used in these tests. Flow distribution characteristics, effects of coolant jetting, and the areas where flow cavitation might occur will be studied by inserting proper instrumentation in the model of the assembly. The inlet and outlet nozzles, the areas around the eggcrate grids, and the strongback support rods will receive particular attention. The study should lead to the elimination of spots of either too low or too turbulent flow and should produce a design with as uniform a flow pattern as practicable.

Coolant mixing is desired in an LMFBR to reduce radial temperature gradients in coolant and cladding. Tests will be conducted to determine the degree of coolant mixing which occurs at spacer grids and between areas which have different axial velocities. These measurements will be made for single and two-phase flow. Tests also will be made to determine the ratio of eddy diffusivity of heat transfer to the eddy diffusivity of momentum transfer in fuel rod bundles.

4.5.4 Relationship to LMFBR Program Plan

The program is related to the fuel assembly design program as described in Volume 6 of the LMFBR Program Plan given below. It differs in scope in that a comprehensive study of the open lattice assembly is required and an evaluation of many assembly geometries and fuel rod arrangements is recommended. The fuel assemblies for the FFTF are canned and without strongback supports; for this reason, the 1000 MWe LMFBR programs should include the development of the strongback concept.

Task 6-1.2 Core and Blanket Design Synthesis

6-1.2.2 Determination of Design Limitation for Core Synthesis

Task 6-2.3 Fuel-Element Support and Spacing

6-2.3.1 Analytical Evaluation of Fuel-Element Support Systems

6-2.3.2 Isothermal Hydraulic Testing of Fuel-Element Support Systems

4.5.5 Major Facilities and Equipment Required

Computing Facility (Tasks 4.5.D and 4.5.E)

A CDC-6600 digital computer and peripheral equipment, or equivalent, is required.

Fast Reactor Facility (Tasks 4.5.A and 4.5.C)

A facility similar to the Demonstration Plant or FFTF is needed to test, under full power conditions, the fuel assembly designs and materials for the 1000 MWe LMFBR.

Film Coefficient Test Facility (Task 4.5.B)

- a. A small sodium loop, similar to the proposed Westinghouse TF-2 loop (see Appendix) with a capacity of about 4 gpm, should be used to design test sections for larger sodium loops.
- b. A larger loop is needed to test, at high velocities, bigger rod clusters up to and exceeding assembly size. A Westinghouse sodium loop GPL-2 (see Appendix) has a capacity of 2000 gpm and could be used for this purpose.

Hydraulic Test Facility (Tasks 4.5.D and 4.5.E)

- a. A small water loop is needed to design test sections for bigger loops and to test small rod clusters. A loop, such as the Westinghouse TF-1 (see Appendix) which has a capacity of 80 gpm, would be acceptable.
- b. A large water loop to do hydraulic and vibrational testing, determine velocity profiles, and perform crossflow experiments is needed. The Westinghouse loop H-1 (see Appendix) has a capacity of 8000 gpm and could be used for these experiments.

Vibration and Hydraulic Resistance Test Facility (Tasks 4.5.D and 4.5.E)

A general purpose sodium loop similar to Westinghouse GPL-1 (see Appendix) is needed to study the behavior of various sizes of rod clusters, depending upon the desired velocity, the pitch, the nature of support system, and the overall hydraulic resistance. The GPL-1 has a capacity of 200 gpm and 125 psi maximum pressure drop.

4.5.6 Program Schedule

The schedule for the development of the fuel assembly for the core is shown in Figure 4.5-1. The schedule assumes that a Demonstration Plant will be available for operational testing of selected assembly designs prior to the construction of the 1000 MWe LMFBR. The major milestones of the program, as shown in Figure 4.5-1, are defined below.

Task 4.5.A - Determination of Thermal Bowing and Thermal Stability of an Assembly

1. Design assembly instrumentation.
2. Insert into Demonstration Plant.
3. Remove from the reactor and examine.

Task 4.5.B - Determination of Film Coefficient Correlation

1. Design test rig for the small flow loop.
2. Commence preliminary tests.

3. Conclude tests.
4. Design test rig for the large loop.
5. Commence tests in the large loop.
6. Conclude tests and analysis.

Task 4.5.C - Evaluation of Strongback Performance in Large Thermal Gradients

1. Design FFTF assembly with strongbacks.
2. Insert into the reactor.
3. Remove from reactor and examine.
4. Design strongback assembly for Demonstration Plant.
5. Insert into the reactor.
6. Remove from the reactor and examine.

Task 4.5.D - Tests of Vibration and Hydraulic Resistance

1. Design test rig for the small flow loop.
2. Commence loop tests.
3. Conclude loop tests.
4. Commence large loop tests of Demonstration Plant assembly.
5. Conclude tests.
6. Commence large loop tests of LMFBR assembly.
7. Conclude tests.

Task 4.5.E - Study of Velocity Profiles and Cavitation

1. Commence loop tests.
2. Conclude loop tests.

Figure 4.5-1. Development of Fuel Assembly Program Schedule

4.6 Development of a Radial Blanket Assembly

4.6.1 Objective

To develop a radial blanket assembly with satisfactory dimensional stability and cooling, to accommodate the sharp change in flux across the radial blanket. The Westinghouse design of the LMFBR requires that a radial blanket assembly be developed which will have dimensional stability in a thermal gradient of 300°F (150°C). Inward bowing of the radial blanket will be minimized to avoid an increase in reactivity due to resulting core compaction. Because of Pu buildup in the blanket assemblies facing the core, an effective orificing system will be provided to allow proper thermal utilization of the blanket throughout its life.

4.6.2 Program Summary

The four tasks which constitute this program deal with testing the performance of rod wrappers or baffles for mixing of flow in the assemblies, the effectiveness of orificing for flow distribution between the various regions of the blanket, and the stability of the rod support system in preventing blanket fuel rod deflection and vibration. The temperature distribution in blanket fuel rods and assemblies also will be measured. The tasks are described in the next section.

4.6.3 Task Description

Task 4.6.A - Tests of Flow Deflecting Baffles

The objective of this Task is to test several designs of devices intended for promotion of coolant mixing in the radial blanket assembly so that one of these can be selected for use in the 1000 MWe LMFBR. The Westinghouse design of a radial blanket assembly is based on the canned assembly concept. The effective mixing of the coolant in the assembly can be increased by including in the design, flow deflecting rod wrappers or baffles. Several such designs will be tested by placing them in the Demonstration Plant and operating under representative reactor conditions.

Then the design with least pressure drop and highest power output can be chosen for use in the 1000 MWe LMFBR.

Task 4.6.B - Tests of Orificing and Power Leveling

The objective of this Task is to test the radial blanket orificing schemes. Orificing of the assemblies in the radial blanket, together with the canned assembly design, will produce a temperature rise of the coolant passing through the radial blanket similar to that of the coolant passing through the core. Thus, no cold coolant will bypass the core, and the thermal efficiency of the reactor will be increased. The orificing in the blanket will also take care of the power gradient in the blanket. This gradient is a function of the Pu buildup in the blanket assemblies facing the core and poses operational problems. An orificing and flow diverting system will be tested under reactor operating conditions in the Demonstration Plant to assure that an effective system is developed capable of handling the large radial power shifts occurring during the life of the blanket as a result of Pu buildup.

Methods will be developed for reducing the sharp temperature gradient at the interface of the core and radial blanket. A method has been proposed which diverts radial flow from the outer edge of the core to the radial blanket. If successful, this method, called "Self-Orificing", would reduce the temperature gradient by increasing the crossflow and turbulent component of velocity. The self-orificing concept can be used also for core and blanket concepts which use perforated cans. The self-orificing concept will be tested in flow loops, then tested in the Demonstration Plant.

Task 4.6.C - Limitation of Blanket Fuel Rod Deflection and Vibration

The objective of this Task is to develop and test the radial blanket rod support system. The size of the rods in the radial blanket assemblies, larger than those in the core assemblies, necessitates development of a different rod support system. The support system will incorporate rod deflection under the action of hydraulic and thermal forces and will keep the rod vibration within reasonable limits. The system developed will be

tested externally prior to operation in test loops. Information of its performance in the Demonstration Plant will provide data on its suitability for use in the 1000 MWe LMFBR.

Task 4.6.D - Determination of Temperature Distribution in Blanket Fuel
Rods and Assemblies

The objective of this Task is to measure the azimuthal temperature variations which will exist in the fuel rods of the radial blanket assembly because of the larger size of the rods and smaller spacing between them. Experimental testing and measurements will be performed to validate the design selected for the 1000 MWe LMFBR in support of the calculations of the temperature distribution.

4.6.4 Relationship to LMFBR Program Plan

From the thermal, hydraulic, and mechanical requirements of development already outlined, the program presented herein is akin to the "Technology of Fuel Elements and Assembly Design" program (LMFBR Program Plan, Task Area 6-2). Development of orificing is presented here and not under the core fuel assembly program. Work under this part of the program is specifically related to the LMFBR Program Plan as follows:

LMFBR Program Plan, Volume 6

Task 6-1.2 Core and Blanket Design Synthesis

6-1.2.3 Determination of Functional Requirements for Blanket
Systems

Task 6-3.1 Blanket Design Technology

6-3.1.1 Evaluation of Design Alternatives

6-3.1.2 Coordination with Fuels and Materials, Physics, and
Sodium Technology Elements

6-3.1.3 Ex-Reactor Proof Testing

4.6.5 Major Facilities and Equipment Required

Computing Facility (Tasks 4.6.B, 4.6.C, and 4.6.D)

CDC-6600 digital computer and peripheral equipment, or equivalent.

Flow Test Facility (Task 4.6.A)

A small water loop is needed to design test sections for bigger loops and to test small rod clusters. A loop, such as the Westinghouse TF-1 (see Appendix) which has a capacity of 80 gpm, would be acceptable.

Flow and Vibration Test Facility (Tasks 4.6.B and 4.6.C)

A large water loop to do hydraulic and vibrational testing, determine the velocity profiles, study self-orificing, and perform crossflow experiments is needed. The Westinghouse loop H-1 (see Appendix) has a capacity of 8000 gpm and could be utilized for these experiments.

Heat Transfer Facility (Task 4.6.D)

A small sodium loop, similar to the proposed Westinghouse TP-2 loop (see Appendix) with a capacity of 4 gpm could be used to design a test section for larger sodium loops.

Flow and Heat Transfer Facility (Tasks 4.6.C and 4.6.D)

A general purpose sodium loop for studying corrosion, heat transfer, vibration phenomena, and mechanical integrity is required. The Westinghouse GPL-1 (see Appendix) could be employed to carry out these types of experiments for various sizes of rod clusters depending upon the desired velocity, the pitch, the nature of the system, and the overall hydraulic resistance. GPL-1 has a capacity of 200 gpm at 125 psi.

Fuel Assembly Test Facility (Tasks 4.6.A and 4.6.B)

Larger loops are needed to test at high velocities, bigger rod clusters up to and exceeding assembly size. A Westinghouse sodium loop GPL-2 (see Appendix) has a capacity of 2000 gpm and would be suitable for that purpose.

Fast Reactor Facility (Tasks 4.6.A, 4.6.B, and 4.6.C)

Fuel rod and assembly configuration previously tested in test loops should be tested in a fast reactor, such as the Demonstration Plant, under realistic full power conditions.

4.6.6 Program Schedule

The schedule for the development of the radial blanket assembly, in time, parallels that of the fuel assembly. The major milestones of the program, as shown in Figure 4.6-1, are defined below.

Task 4.6.A - Tests of Flow Deflecting Baffles

1. Design test rig for flow loop.
2. Commence tests.
3. Conclude out-of-core tests.
4. Insert assembly in Demonstration Plant.
5. Conclude in-core tests.

Task 4.6.B - Tests of Orificing and Power Leveling

1. Construct scale model of the core and blanket.
2. Insert in the flow loop.
3. Conclude flow tests.
4. Commence flow measurements in Demonstration Plant.
5. Conclude flow measurements in Demonstration Plant.

Task 4.6.C - Limitation of Fuel Rod Deflection and Vibration

1. Design and build test rig.
2. Commence loop tests.
3. Conclude loop tests.
4. Commence tests in Demonstration Plant.
5. Conclude tests in Demonstration Plant.

Task 4.6.D - Determination of Temperature Distribution in Rods and Assembly

1. Design and construct test rig.

2. Commence tests in a flow loop.
3. Conclude tests in a flow loop.
4. Insert into Demonstration Plant.
5. Conclude in-core tests and analysis.

Figure 4.6-1. Development of a Radial Blanket Assembly Program Schedule

4.7 Development of Control Rods and Control Rod Drives

4.7.1 Objective

To develop control rod drives which are jam free in operation. Long-term freedom from deposition of fission products, Na freeze-out, or effects of mass transfer is required. The performance of the control rod blades must not be impaired by dimensional distortion, attack by fission products, or erosion by mass transfer in Na. The size of steps in control rod movement should be limited to 1.2¢/sec reactivity increase on rod withdrawal. Rod insertion should not take longer than 1.5 seconds.

4.7.2 Program Summary

The five tasks of this program deal with the determination of the coefficients of sliding friction in Na, studies of the need for internal cooling of the rods, and the effect of deposition of volatile fission products in the drive mechanisms. Measurements of reactivity worth of control rods and the change in the reactivity worth with the burnup condition of the core are also included. The tasks are described in the next section.

4.7.3 Task Description

Task 4.7.A - Measurements of Coefficients of Sliding Friction

The objective of this Task is to measure the coefficients of friction of the materials and components in the control rod drive mechanism and control rods. The successful design of control rod drives and control rod guide tubes in the fuel assemblies requires a better knowledge of the coefficients of sliding friction of the materials in question in a hot sodium environment. The coefficients of sliding friction will be measured over the whole temperature-range, from just molten sodium up to 20% over-power temperature condition of the coolant. Knowledge of the friction phenomenon has a bearing on the allowable clearances which can be permitted in the control rod guide tubes, as well as the possibility of inducing galling of the surfaces in contact. The materials to be tested are the interfaces of Ta against 316 SS or 316 SS against 316 SS, should

B₄C cladding in stainless steel be chosen as the control material. The influence of impurities in Na on the coefficients of friction also will be determined.

Task 4.7.B - Study of the Effect of Deposition of Fission Products

The objective of this Task is to study the behavior of the control rod drive when contaminated by cover gas and fission products. Although the present Westinghouse design of control rod drives provides for continuous argon gas purging of control rod drive mechanisms, the effect of the failure of the gas purge mechanism might be the introduction of volatile fission products into the housings. The vented fuel design, of course, would lead to their presence in the bulk coolant and in the gas space directly above it. To estimate the effect of introduction of these volatile fission products into the drive housings and their possible precipitation there, realistic tests in an operating reactor will be carried out. This will be done by operating one of the control rod drives in the FFTF or the Demonstration Plant with the purge gas turned off. The information on the effect on operation of the drives then would be directly applicable to the 1000 MWe LMFBR.

Task 4.7.C - Study of the Need for Internal Cooling of Control Rods

The objective of this Task is to study the need for internal cooling of the control rods, to develop coolant orifices for the control rods, and to study the need for control rod followers. Trouble-free operation of Ta control rods will require that they be internally cooled to remove the energy deposited in them as a result of neutron capture and gamma heating. Because of low neutron capture cross sections of the control rod materials in the neutron spectrum characteristic of the LMFBR, either considerable thickness has to be given to control rods to assure sufficient neutron absorbing worth, or a large number of rods has to be provided in the design. The latter alternative is economically unattractive due to high costs of control rod drives. As a result of the above considerations, cooling channels may be required in the control rods. The final decision will be made after measuring the amount of thermal distortion in

Ta control rods operating with and without internal cooling channels in a representative environment with high gamma flux gradient. Orifices for control rods will be developed to insure that adequate cooling occurs at any rod position. Equally important, control rods should not be over-cooled, causing needless bypass. Control rod followers also will be investigated.

Task 4.7.D - Determination of the Reactivity Worth of Control Rods

The objective of this Task is to measure the control rod worths as functions of position and burnup. At present, considerable weaknesses exist in the knowledge of the high energy neutron capture cross sections of control rod materials, such as Ta, and in the ability to calculate space-dependent neutron energy spectra. For this reason, the predictions of the reactivity worth of control rods located at various radial and axial positions in the core may be in considerable error, particularly with respect to control rods located in different enrichment zones. To reduce this margin of error, measurements of the reactivity worth of control rods in zones of different Pu fraction will be made in a critical facility, such as the ZPPR, and confirmed in the Demonstration Plant to check the calculational predictions. In the same way, the effects of burnup on the worth of a control rod located at a fixed position will be measured. In this instance, the measurements will be made in an operating reactor because of the difficulty of preparing a critical experiment which models accurately the isotopic composition of the core representative of fission products and heavy metal isotopes at high burnup. This task will be coordinated with Task 4.8.D.

Task 4.7.E - Evaluation of Changes in Control Rod Worth at Core Edge

The objective of this Task is to measure the reactivity worth of control rods which are near the blanket as a function of Pu buildup between refueling. One of the effects of breeding in an LMFBR is the buildup of Pu inventory in the radial blanket assemblies which face the core. Consequently, the reactivity worth of control rods located near the edge of the core is expected to be a strong function of residence time of the

adjacent blanket assemblies. The accuracy of the prediction of this effect by means of calculations is questionable and will be checked by careful measurements of the reactivity worth of peripheral control rods at the beginning and at the end of the refueling cycle. These measurements will be made in the Demonstration Plant. This task will be coordinated with Tasks 4.7.D and 4.8.D.

4.7.4 Relationship to LMFBR Program

The work performed under AEC Contract (30-1)-3684 (Ref. WARD-30) and under a Westinghouse-funded general research program covers the development of the control rod drive mechanism. Work under this program is related to Task 6-3.2, "Control and Safety Assembly Design Technology" in Volume 6 of the LMFBR Program Plan and Task Area 3-10, "Reactivity Control Drives" in Volume 3.

4.7.5 Facilities and Equipment Required

Computing Facility (Tasks 4.7.C, 4.7.D, and 4.7.E)

Digital computer, CDC-6600 and peripheral equipment, or equivalent.

Coefficient of Friction Test Facility (Task 4.7.A)

A sodium test loop with the capability of accommodating a device for measurement of the coefficients of static and dynamic friction between sets of surfaces is required. Conditions of static and flowing sodium at temperature ranges up to 650°C (1200°F) are needed. A loop, similar to Westinghouse TF-2 (see Appendix), is needed.

Control Rod Drive Test Facility (Tasks 4.7.B and 4.7.C)

A facility is needed for development of control rod drive mechanisms which will consist of a sodium test loop capable of accepting a full scale unit cell module consisting of control rod drive mechanism (CRDM), drive line guide tube, integral fuel assembly. A facility, similar to Westinghouse DMT-1 facility (see Appendix), is needed.

Fast Critical Facility (Task 4.7.D)

A critical facility, such as ZPPR, with capability of accommodating zones of different Pu fraction (enrichment) is needed for measurements of reactivity worths of control rods as a function of position and size.

Fast Reactor Facility (Tasks 4.7.D and 4.7.E)

The access to an operating fast reactor is needed for measurements of changes in worths of control rods at various conditions of burnup and Pu buildup in the blanket. The reactor should have the capability of accommodating control rods of various sizes.

4.7.6 Program Schedule

The schedule for the development of the control rods and control rod drives is shown in Figure 4.7-1. The major milestones of the program, as shown in Figure 4.7-1, are defined below.

Task 4.7.A - Measurements of Coefficients of Sliding Friction

1. Design and construct test rig.
2. Commence loop tests.
3. Conclude loop tests.

Task 4.7.B - Study of the Effect of Deposition of Fission Products

1. Commence tests of CRDM.
2. Conclude tests of CRDM.
3. Commence tests in Demonstration Plant.
4. Conclude tests in Demonstration Plant.

Task 4.7.C - Study of the Need for Internal Cooling of Control Rods

1. Commence operation of prototype in Demonstration Plant.
2. Conclude the tests.
3. Remove the control rods and examine for distortion.

Task 4.7.D - Determination of the Reactivity Worth of Control Rods

1. Commence worth measurements in ZPPR.

2. Conclude worth measurements in ZPPR.
3. Commence worth measurements in FFTF.
4. Conclude worth measurements in FFTF.
5. Commence worth measurements in Demonstration Plant.
6. Conclude worth measurements in Demonstration Plant.

Task 4.7.E - Evaluation of Changes in Control Rod Worth at Core Edge

1. Commence measurements in Demonstration Plant.
2. Measure reactivity worth at first refueling.
3. Measure reactivity worth at second refueling.
4. Measure reactivity worth at third refueling.

Figure 4.7-1. Development of Control Rods and Drives Program Schedule

4.8 Studies of Modular Reactor Geometry

4.8.1 Objective

To better understand the behavior of a modular fast reactor by means of analysis and experiment. For successful development of the modular reactor geometry, analytical methods will be developed for treating explicitly and in sufficient detail the space-dependent kinetics problem. The models developed will be checked by the results of critical experiments. In particular, the objective is to develop methods for predicting the magnitude of neutronic coupling between the core modules, the determination of time constants of the reactivity worth of control rods and Na voids in adjacent modules.

4.8.2 Program Summary

The four tasks which constitute this program cover:

- The measurement of transfer functions in modular geometries
- Development of analytical methods
- Testing of analytical modules in multi-modular critical experiments
- Measurements of control rod worths in modular geometries

The tasks are described in the next section.

4.8.3 Task Description

Task 4.8.A - Measurement of Transfer Functions in Modular Geometry

The objective of this Task is to study, develop and then compare two different methods to measure the transfer function of the modular reactor. To place the design of the modular reactor concept on a firm basis with regard to kinetic behavior, the methods of measuring transfer functions in a modular reactor with a fast neutron spectrum will be developed. In principle, either oscillator methods or the noise analysis techniques may be adapted for use in modular geometries. In the oscillator method, reactivity would be varied in one module, and then the kinetic response of other modules and the reactor as a whole will be determined. Recent developments in more efficient data processing equipment and the

improvement in computational techniques suggest that the cross correlation technique of noise analysis, using two detectors to eliminate the uncorrelated noise, may provide the most promising approach. The two methods will be studied. An additional objective will be to provide the tools and computational techniques for reducing experimental data obtained in measurements of transfer functions in modular geometries for a wide range of impressed frequencies and to explore the possibilities of kinetic instability.

Task 4.8.B - Analysis of Space-Time Kinetics

The objective of this Task is to develop the analytical tools needed for kinetics and stability analyses of the modular reactor. At present, no adequate mathematical model exists for treating the kinetic coupling between modules of a multi-core reactor. Existing models are limited to thermal cores of the seed-and-blanket type (Shippingport) or fuel separated by void (safe geometries for fuel processing). Fast systems treated are limited to weapons-type small metallic assemblies. A kinetic model, based on modal-nodal (or quasistatic) approaches, will be developed for modular cores. The stability criteria for multi-modular fast reactors will be developed. The intention of the Task is to replace the point kinetics model; its applicability to reactors in modular geometry is highly questionable.

A multi-dimensional, multi-group mathematical model will be developed for predicting the reactor transfer functions in modular geometries. The model will have the capability of treating the accident conditions, with partial core meltdown or disassembly having taken place.

Task 4.8.C - Critical Experiments in Modular Geometry

The objective of this Task is to perform critical experiments to study module-module coupling, stability, and kinetics. Critical experiments will be carried out in modular geometry to substantiate the calculational predictions of various physics parameters. The modular geometry will consist of at least two modules, and preferably three or four. The core modules will be varied in different experiments from one containing

only depleted U, to one containing Pu concentrations corresponding to the end of cycle blanket composition. A variation in blanket thickness separating the modules will also be introduced. The objective of these tests is to establish the dependence of the kinetic coupling between the modules on the composition and size of the internal blanket. The effect of the amount of neutronic coupling on kinetic stability and ease of control of the reactor, particularly at the end of cycle, will be studied in detail, and stability criteria will be established. These studies will be carried out with critical assemblies smaller than a full-size mockup of 1000 MWe LMFBR, in the ZPPR or one of the ZPR's, using the methods developed in Task 4.8.A.

Task 4.8.D - Measurement of Control Rod and Void Interaction in Modules

The objective of this Task is to measure the reactivity worth of control rods and voids to determine the interaction between modules of the control rods or voids in the several modules. Critical experiments in modular geometries will be used for measurement of control rod interactions at adjacent positions in a module and at symmetrical positions in different modules. The effect of changing the thickness of the blanket separating the modules on the reactivity worth of the control rods will be measured. In addition, the effects of voiding Na from a central region and voiding of a complete core and axial blanket of one module will be measured. This measurement will show whether the modular concept of core effect on reactivity is correct or not. The control rod worth and void interactions will be measured in zero power, full size critical mockups of the 1000 MWe LMFBR in ZPPR or the ZPR's. This task will be coordinated with the work in Task 4.7.D.

4.8.4 Relationship to LMFBR Program

The LMFBR Program Plan does not refer directly to modular reactor cores. The plan, however, in Task Area 9-3, "Theory and Computation for Reactor Analysis and Design", does refer to the problems associated with the prediction of space-dependent kinetics in large reactors.

Other references in the LMFBR Program Plant are Task Area 6-1, "Reactor Core Engineering" in Volume 6 and Task 10-2.2, "Codes for Analysis of Hypothetical Accidents", in Volume 10.

4.8.5 Major Facilities and Equipment Required

Computing Facility (Tasks 4.8.A, 4.8.B, and 4.8.C)

A CDC-6600 digital computer plus peripheral equipment, or equivalent, is needed.

Thermal Critical Facility (Task 4.8.A)

The methods of measurement of transfer functions in modular cores may be initially tested in a thermal critical facility, such as Westinghouse CRX facility (see Appendix), which has the capability of accommodating modular reactors.

Fast Critical Facility (Tasks 4.8.A, 4.8.B, and 4.8.D)

A fast critical facility is needed for measurement of neutronic coupling between modules of all-metal reactors. Experiment volumes of at least 3 m^3 , and up to 7 m^3 may be required. Non-accident kinetic and control rod worth measurements can be taken in dry critical assemblies similar to ZPR and ZPPR.

4.8.6 Program Schedule

The schedule of the nuclear characteristics studies of the modular reactor geometry is given in Figure 4.8-1. Major milestones of the program, as shown in Figure 4.8-1, are defined below.

Task 4.8.A - Measurement of Transfer Functions in Modular Geometry

1. Preliminary methods tests in a thermal critical facility.
2. Conclude preliminary tests.
3. Commence tests in a fast critical facility.
4. Conclude tests in a fast critical facility.
5. Conclude comparison with analytical methods.

Task 4.8.B - Analysis of Space-Time Kinetics

1. Commence analytical investigations.
2. Commence preparation of computer program.
3. Conclude preparation of computer program.

Task 4.8.C - Critical Experiments in Modular Geometry

1. Commence tests of the effect of blanket thickness.
2. Commence tests of the effect of blanket composition.
3. Conclude tests.

Task 4.8.D - Control Rod and Void Interaction in Modules

1. Commence measurements of rod worth and interaction in modules.
2. Commence measurements of void interaction in modules.
3. Conclude tests.

Figure 4.8-1. Studies of Modular Reactor Geometry Program Schedule

4.9 Development of Reactor Vessel Internals

4.9.1 Objective

To develop the internal reactor structure for the 1000 MWe LMFBR that has a design life of thirty years and is free from vibrations which might lead to core damage or flow maldistribution. Easy accessibility for maintenance and inspection is required. The vessel internals also must assure appropriate Na velocity profile at core inlet and prevent the drawing of the coolant from the core by a siphon effect. Vortex formation in the coolant surface above the core also must be suppressed.

4.9.2 Program Summary

The tasks of this program deal with tests of scale models of the core and internals, development of an effective anti-siphon device to prevent core drainage, measurements of displacement and vibration frequencies in the Demonstration Plant, and the development of the vortex suppression plate. The tasks are described in the next section.

4.9.3 Task Description

Task 4.9.A - Studies of Coolant Flow Distribution in Reactor Vessel

The objective of this Task is to determine experimentally the distribution of coolant flow within the reactor vessel. Due to the difficulties in accurate predictions of distribution of coolant flow through a complicated reactor structure, a scale model of the reactor core, core support plate, and the flow baffles will be built and tested extensively. The model will take full account of the complexities in the flow pattern introduced by the variable orificing of the radial blanket assemblies and the flow diverting structure needed to assure even distribution of coolant between reactor modules. The in-model instrumentation will provide for the measurement, at a sufficient number of points, of the coolant flow distribution, hydraulic resistance, and the vibration characteristics of the structure. An option will be provided for testing an alternate, non-modular reactor structure.

Task 4.9.B - Development of Coolant Siphon Breaker

The objective of this Task is to design, develop, and test two siphon breaker designs, one of which will be used in the 1000 MWe LMFBR. A break in the main coolant pipe, leading to the drainage of coolant from the reactor and blankets, would lead to the melting of the reactor core. The drainage could be effected by a siphon effect, should no provisions be made for its prevention. The siphoning of the coolant from the reactor vessel can be prevented, however, by either placing all piping on a level above the reactor vessel or by including in the coolant loops an effective siphon breaker. The latter option was chosen for the Westinghouse concept of the 1000 MWe LMFBR, but the design of the device has not progressed beyond the conceptual stage. A reference and an alternate design of a siphon breaker will be developed, and scale models will be tested under a variety of conditions including pressure surges, partial blockage of inlet pipes, main coolant pipe break, etc. The siphon breaker with the best performance will be selected for implementation in the LMFBR.

Task 4.9.C - Measurements of Temperature Distribution, Displacement, and Vibrations of Vessel Internals in the Demonstration Plant

The objective of this Task is to measure the temperature distribution, displacements, and vibrations in the Demonstration Plant reactor internals. The accuracy of the methods used in designing the core internals will be verified by placing detailed instrumentation in the core structure of the Westinghouse Demonstration Plant. The instrumentation will verify the distribution of temperatures in the vessel internals, their displacements under the action of thermal gradients, and the amount of vibration induced by coolant flow. While it is recognized that the design of the vessel internals in the Demonstration Plant will differ considerably from that of the 1000 MWe LMFBR, the problem of thermal gradients and hydraulic resistance will be quite similar. The Demonstration Plant will provide an excellent opportunity to test the design methods and to indicate, if needed, design changes.

Task 4.9.D - Determination of the Effect of Vortex Suppressor on Pressure Drop

The objective of this Task is to determine experimentally the effects of the vortex suppressor on the reactor pressure drop and flow distribution. Preliminary studies indicate the need of including in the design of the vessel internals a vortex suppressor. The purpose of this device is to suppress the coolant vortex which might form in the interface between the coolant and the cover gas in the space above the core, so as to avoid the problems which existed in the initial design of the Dounreay reactor. The presence of the vortex suppressor plate at this location is expected to increase the pressure drop in the coolant as it passes through the core. Studies in scale models of the core and the vessel internals will be performed to determine the effect of the location of the vortex suppressor plate on the pressure drop and the flow distribution in the reactor.

4.9.4 Relationship to LMFBR Program

The work under this program is related to and includes some of the work specified in Volume 6 of the LMFBR Program Plan, Task 6-1.3, "Core Support Structure" and Task 6-3.3, "Core Structure Design Technology".

4.9.5 Major Facilities and Equipment Required

Computing Facility (Tasks 4.9.A, 4.9.C, and 4.9.D)

A digital computer, CDC-6600 plus peripheral equipment, or equivalent, would be required.

Reactor Flow Model Test Facility (Tasks 4.9.A, 4.9.B, and 4.9.D)

A mission oriented flow loop is required to test scale models of reactor cores and vessel internals for flow distribution and vibration phenomena under conditions of geometric, dynamic, and vibrational similarity.

Some of this work could be accomplished with a large water loop, such as Westinghouse H-1 (see Appendix). Other experiments would require a sodium

loop with flow requirements an order of magnitude greater than that of the Westinghouse GPL-2 loop (see Appendix).

Fast Reactor Facility (Task 4.9.C)

Access to a fast reactor facility, such as the Demonstration Plant, will be needed for measurements of temperature distribution, displacements, and vibrations in reactor internals.

4.9.6 Program Schedule

The schedule for the development of reactor vessel internals is given in Figure 4.9-1. The major milestones of the program, as shown in Figure 4.9-1, are given below.

Task 4.9.A - Studies of Coolant Flow Distribution in Reactor Vessel

1. Design and construct scale model of one module.
2. Commence loop tests.
3. Conclude tests.
4. Design and construct scale model of the reactor core.
5. Commence loop tests.
6. Conclude tests.

Task 4.9.B - Development of Coolant Siphon Breaker

1. Design and construct scale model of siphon breaker.
2. Commence loop tests.
3. Conclude loop tests.

Task 4.9.C - Measurements of Temperature Distributions, Displacements, and Vibrations of Vessel Internals in Demonstration Plant

1. Design test instrumentation for insertion into Demonstration Plant.
2. Commence tests in the Demonstration Plant.
3. Conclude tests.

Task 4.9.D - Determination of the Effect of Vortex Suppressor on Pressure Drop

1. Design and construct scale model of the vortex suppressor.

2. Commence loop tests.
3. Conclude loop tests.

Figure 4.9-1. Development of Reactor Vessel Internals Program Schedule

4.10 Development of Computer-Assisted Fuel Management System

4.10.1 Objective

To develop a computer-assisted fuel management system. A system will be developed to carry out continuous assessment of isotopic inventory of the core and the record of the total energy generated to optimize the economic operation of the fuel management in the 1000 MWe LMFBR. The system will be capable of identifying the burnup condition of each fuel assembly, its isotopic inventory, current value of the breeding ratio, and indicating which assemblies should be refueled and which recycled. The system will be based on in-core detectors, analog-to-digital converter, and a digital computer to carry out the recording-predicting function.

4.10.2 Program Summary

Starting with the economic evaluation of the system the five tasks of this program cover the study of advantages of canned fuel assemblies for keeping a record of energy release through the development of in-core detectors for sensing power distributions and the local U/Pu ratio. A test of the system is proposed for installation in the Demonstration Plant. The tasks are described in the next section.

4.10.3 Task Description

Task 4.10.A - Evaluation of Economic Advantages of Continuous Fuel Management System

The objective of this Task is to determine the need for a computer-assisted fuel management schedule for the 1000 MWe LMFBR. The development of the computer-assisted fuel management system will be preceded by a study of the costs involved in installing the system of in-core sensors, the analog-to-digital converter, and the digital computer. These costs will be compared with the gains that may be possible in the economic operation of the plant, and the aggregate of breeder reactors as a whole, resulting from the ability to optimize the fuel management in operation.

This economic analysis will lead to either rejection or adoption of the computer-assisted fuel management system. In the latter case, the Tasks 4.10.B through 4.10.E will be carried out.

Task 4.10.B - Comparison of Canned versus Uncanned Fuel Assembly

The objective of this Task is to compare the canned and uncanned fuel assemblies, from the standpoint of instrumentation capabilities, as well as safety. The choice of canned versus uncanned fuel assembly, in addition to its effect on void propagation, will also be considered with regard to the measurement of its energy release. Such measurement can be performed in canned assemblies by inserting thermocouples at the inlet and outlet of the coolant to each assembly. The temperature difference in the coolant plus the knowledge of the coolant flow rate not only gives the record of power generated but also the capability of locating fuel assemblies which are overheating through flow blockage or some other reason. These bookkeeping and control advantages will be compared with the advantages of cross flow as it is effective in preventing the local overheating due to flow blockage, while preventing, at the same time, the possibility of locating the hot spot. A tradeoff between these two concepts will be carried out.

Task 4.10.C - Development of In-Core Power Distribution Sensors

The objective of this Task is the developmental testing of in-core power distribution sensors. Withdrawal of control rods from the core and the breeding of Pu in the axial blankets, which occurs during the operating cycle, lead to considerable changes in the axial power profile in the core. Knowledge of these changes is of interest because of their relation to the location of the hot spot in each core, as well as for the information they supply on the Pu inventory in the axial blankets and its management. Three schemes for sensing axial power distributions are under consideration: (1) flux wires (similar to those used in the Yankee reactor), (2) aeroballs (flux sensors loaded and removed pneumatically), and (3) the miniature fission chambers. None of these systems have been tested adequately in fast power reactors. Such tests will be performed

simultaneously with the development of the flux-sensing materials resistant to high temperature and flux. Reliable and reproducible readout methods also will be developed. Based on these tests, preferably carried out in the FFTF or the Demonstration Plant, an optimum system will be selected.

Task 4.10.D - Development of Methods for Sensing U/Pu Ratio

The objective of this Task is the development of a technique for monitoring the fuel U/Pu ratio. An effective fuel cycle management requires the possibility of frequent sensing of the burnup and breeding status of the fuel while this fuel undergoes irradiation. Thus, the information on its status at refueling time will be influential in determining the future location of the fuel. For this purpose, a non-destructive method will be developed for sensing the changes in the U/Pu ratio in each fuel assembly. Either removable fuel sections or insertable devices containing fuel samples of proper U/Pu ratio will be developed to sense the changes in the inventory of heavy metal isotopes. A differential γ -ray or neutron spectroscopy method will be selected for automatic analysis at the plant. The results will be fed and stored in the memory of the fuel management computer. Selection of the number and location of sampling thimbles, the method of removing the samples, the frequency of sampling, and the computer software for converting the results of analysis into heavy metal inventories, etc., will be developed.

Task 4.10.E - Test of the Fuel Management System in the Demonstration Plant

The objective of this Task is to test the fuel management system in the Demonstration Plant by comparison with destructive fuel assays. Once the in-core fuel status sensing system is developed, and the methods of integrating the information from the thermocouples, flow meters, burnup sensors, flux wires, etc. are obtained, the system will be tested by installing it in the Demonstration Plant. As the fuel is unloaded from the plant at the end of each fuel cycle, it will be destructively sampled and subjected to mass spectroscopic analysis. The heavy metal inventories

and the burnup status then obtained will be used to check and improve the prediction of the on-line fuel management system. Thus, the plant operator will be supplied with the complete information on the fuel status by assembly, and at all times, in the core life. He will be able, then, to improve the economics and eliminate the expense of destructive isotopic analysis of the spent fuel as well as the delay which is entailed in feeding this information into the fuel management. On the basis of the performance of the system in the Demonstration Plant, the system for the 1000 MWe LMFBR will be designed.

4.10.4 Relationship to LMFBR Program

The following work was suggested by the results from Tasks I and II of the 1000 MWe Follow-on Study. The LMFBR Program Plan has the following relevant tasks: Subtask 6-1.2.4, "Evaluation of Fuel Management Schemes" in Volume 6 and Subtask 9-3.4.3, "Improvements of Fuel-Cycle and Management Schemes" in Volume 9.

The computer-assisted fuel management, however, is not discussed explicitly.

4.10.5 Facilities and Equipment Required

Computing Facility (Tasks 4.10.D and 4.10.E)

A CDC-6600 digital computer and peripheral equipment, or equivalent, is needed.

Non-Destructive U/Pu Assaying Facility (Task 4.10.D)

A laboratory for non-destructive assaying of U/Pu ratio is needed. It should be equipped with instrumentation for X-ray spectroscopy and/or a generator of monoenergetic neutrons of about 150 keV energy.

Destructive U/Pu Assaying Facility (Tasks 4.10.C and 4.10.E)

A laboratory equipped for destructive analysis of fuel for the determination of U/Pu ratio is needed. The facility should be equipped in hot cells and the mass spectrometer, i.e. should be similar to the Westinghouse PIF and ASL facilities (see Appendix).

Fast Irradiation Test Facility (Task 4.10.C)

Irradiation thimble in a fast flux test facility, such as FFTF, is needed for irradiating in the fast spectrum of U/Pu ratio sensing materials.

Demonstration Plant Facility (Task 4.10.E)

A Demonstration Plant is needed to provide an opportunity for installing the computer-assisted fuel management system and for providing fuel which can be destructively analyzed to test the predictions of the fuel management system.

On-Line Computer Facility (Task 4.10.E)

A digital computer must be specified for use in the Demonstration Plant to handle the continuous fuel management system. The computer must incorporate an analogue-to-digital connector, adequate memory storage capacity, and printing plus graphical or visual display devices.

4.10.6 Program Schedule

The schedule for the development of the computer-assisted fuel management scheme is given in Figure 4.10-1. The major milestones of the program, as shown in Figure 4.10-1, are given below.

Task 4.10.A - Evaluation of Economic Advantage of Continuous Fuel Management System

1. Commence economic evaluation.
2. Conclude economic evaluation.

Task 4.10.B - Comparison of Canned versus Uncanned Fuel Assembly

1. Initiate design study.
2. Conclude design of alternative assemblies.
3. Insert developmental assemblies into Demonstration Plant.
4. Conclude the in-core testing.

Task 4.10.C - Development of In-Core Power Distribution Sensors

1. Select concept and materials.

2. Design and construct test devices.
3. Start irradiation tests in FFTF.
4. Evaluate preliminary irradiation tests.

Task 4.10.D - Development of Methods for Sensing U/Pu Ratio

1. Select promising concept.
2. Design and construct sensing device.
3. Start in-core tests in FFTF.
4. Conclude in-core tests.

Task 4.10.E - Test of Fuel Management System in the Demonstration Plant

1. Specify design parameters of system.
2. Construct and install in the Demonstration Plant.
3. Unload first batch of fuel.
4. Unload second batch of fuel; analyze first batch.
5. Unload third batch of fuel; analyze second batch.
6. Analyze third batch of fuel; assess performance of the system.

Figure 4.10-1. Development of Computer-Assisted Fuel Management System Program Schedule

4.11 Development of a Non-Modular Core

4.11.1 Objective

To develop a non-modular core design which, in addition to good economics, will permit safe operation. Preliminary calculations indicate that the lowest fuel cycle cost is obtainable with a non-modular core geometry.

The development of the non-modular core makes unnecessary the need to develop sophisticated methods of analysis of kinetic interactions between the modules, experimental tests for the effect of the blanket thickness on this coupling, and the investigation in support of developing a hydraulic system which will assure equal distribution of coolant to each of the modules. The work performed in program 4.4 and 4.8 of this report will be closely coordinated with the effort in this program.

4.11.2 Program Summary

The three tasks constituting the program provide for obtaining information specific to the non-modular core. It is assumed that the bulk of the information developed in the previous programs will be applicable to the non-modular design. The three tasks deal with

- Investigation of the mechanism of void formation in large cores
- The probability of void propagation through uncanned assemblies
- The extent of damage and energy releases in case of meltdown

The tasks are described in the next section.

4.11.3 Task Description

Task 4.11.A - Study of the Mechanism of Void Propagation in Large Cores

The objective of this Task is to develop models for the analysis of coolant void formation and propagation. At present, little information exists with regard to the mechanism of coolant void formation in large cores. The sequence of events leading to the propagation of the void is not well understood. In particular, it is not known whether fuel melting would occur in a small volume initially voided, with the disintegrating

pieces of fuel then becoming agents of void propagation in being dispersed, or whether a large volume of the core would be voided first and fuel collapse then occur. Analytical models will be developed to describe the sequence of events and to predict the thermodynamic conditions of the core contents in the event of a catastrophic transient. The maximum void (and its duration) which can be withstood by the core without propagation and collapse will be determined. Preventive and remedial courses of action on the part of the reactor operator will be studied and their feasibility will be evaluated. Similar information is needed for a modular reactor, but it is of greater importance for the design of a non-modular reactor.

Task 4.11.B - Determination of Void Propagation with Uncanned Fuel

The objective of this Task is to compare canned and uncanned fuel assembly designs, from the standpoint of coolant void generation and propagation. The scanty experience with operating fast reactors suggests that the use of canned fuel assemblies may be instrumental in preventing the void propagation throughout the core. Implicit in this line of reasoning is the assumption that the local fuel failure initiating the void formation is non-explosive and does not carry through the fuel cans into the adjacent assemblies, and that such penetration as does occur is due to melt-through and not blow-through. In the case of the uncanned fuel assemblies, initial overheating due to flow blockage may be prevented, but once the void starts to form, there is no mechanical barrier to void propagation such as that provided by the fuel cans. The relative probability of void generation in these two situations will be assessed. A design decision will be made with regard to the fuel assembly design based on the outcome of experiments in test facilities capable of accommodating several fuel assemblies and allowing nuclear transient testing and loss-of-flow testing. This information is similar to that needed for a modular reactor, but it is of greater importance for the design of a non-modular reactor.

Task 4.11.C - Determination of the Effects of Accidents in Modular and Non-Modular Cores

The objective of this Task is the analytical and experimental determination of energy releases from the modular and non-modular reactors so as to determine the relative merits of the two concepts. In order to compare and choose between the modular and the non-modular core geometry, the total energy release and the extent of core damage in the case of a major meltdown will be assessed for each concept. In studies of the voiding of coolant in the modular cores, it has been assumed that the transient and its effects will be confined to one module. This presumes that the blanket provides sufficient nucleonic decoupling between the modules and that it forms an adequate physical barrier against the movement of the coolant and the debris from the melting module. All these assumptions will be re-examined in the study of the energy releases in the reference and the alternate reactor configuration for several different reactor accidents. Prevention of complete plant loss and the possibility of restoring the plant to an operating condition in the event of major core damage are strong functions of the maximum energy release in an accident. Analytical results will be substantiated by experiments in the STF and a SPERT-type facility (ETF).

4.11.4 Relationship to LMFBR Program

The work described in this program relates to Task Area 6-1, "Reactor Core Engineering", in Volume 6 of the LMFBR Program Plan. Aspects of core design are covered which relate specifically to the problems which might accrue from the use of non-modular design of the reactor in place of the modular design. In addition, some of this effort is described in Task 10-2.2, "Codes for Analysis of Hypothetical Accidents", in Volume 10.

4.11.5 Major Facilities and Equipment Required

Computing Facility (Tasks 4.11.A, 4.11.B, and 4.11.C)

Digital computer, CDC-6600 plus peripheral equipment, or equivalent.

Void Propagation Test Facility (Tasks 4.11.A and 4.11.B)

A nuclear test facility of the STF type is needed which will accommodate two or more fuel assemblies in a flowing Na environment and will allow nuclear transients to be used to produce final failure. It should be instrumented for study of void propagation in the assemblies.

Energy Release Test Facility (Tasks 4.11.B and 4.11.C)

The ETF (a SPERT-type facility) is needed for study of the energy releases accompanying the core meltdown in nuclear transients and loss-of-flow transients, as well as for studies of void propagation for canned and uncanned fuel assemblies.

4.11.6 Program Schedule

The schedule for the development of the non-modular core is given in Figure 4.11-1. The major milestones, as shown in Figure 4.11-1, are defined below.

Task 4.11.A - Study of the Mechanism of Void Propagation in Large Cores

1. Formulate mathematical model.
2. Code for the computer.
3. Debug and issue a report.

Task 4.11.B - Determination of Void Propagation with Uncanned Fuel

1. Design and manufacture fuel assemblies.
2. Set up test configuration; commence tests.
3. Conclude and evaluate tests.

Task 4.11.C - Determination of the Effects of Accidents in Modular and Non-Modular Cores

1. Design and manufacture test fuel assemblies.
2. Set up energy release experiment in modular geometry.
3. Set up energy release experiment in non-modular geometry.
4. Conclude experiments.
5. Conclude analysis.

Figure 4.11-1. Development of Non-Modular Core Program Schedule

5. PLANT SYSTEMS AND COMPONENTS

5.1 Reactor Vessel Development

5.1.1 Objective

To develop the technology of sodium reactor vessels such that its design and service specifications will meet the functional requirements of the 1000 MWe LMFBR plant.

5.1.2 Program Summary

The program is divided into four major tasks. The first two tasks are used to develop the design code criteria and stress allowables, and provide for the development of analytical tools. The last two tasks are used to develop and test closures and inservice inspection techniques. Tests will include the operation of prototype units in representative environments and under representative operational conditions. The approach to the development is described in Figure 5.1-1. These Tasks are described in the next section.

5.1.3 Task Descriptions

Task 5.1.A - Development of Design Criteria

The objective of this task is to compile a set of standard design guides for sodium component design. The establishment of a stress criteria and allowable stress values for temperatures in excess of 800°F for the construction material will be emphasized.

The design of a reactor vessel represents only a modest extension of the presently available ASME codes. At present, however, many of the system components are being designed for service at temperatures >1000°F. The accepted section of the ASME code for nuclear component design does not consider operation at temperatures in excess of 800°F. Specifically, extensions of current codes or generation of new codes which are directly applicable to the problem are needed. As a minimum, the code should extend the design criteria into the creep range (>800°F) for both steady-state pressure loading and the combination of creep and low-cycle fatigue. The extension also should clarify the relationship between biaxial and uniaxial creep.

Figure 5.1-1. Model of Reactor Vessel Development Process

The development of this task will be useful and necessary for the design of components such as the IHX, steam generator, and piping, as well as for the design of the reactor vessel itself.

This program will be closely coordinated with the materials development programs. Material testing also should investigate uniaxial and biaxial creep, stress-rupture, ductility, and accumulative fatigue damage.

Task 5.1.B - Development of Analytical Techniques

The objective of the program is to establish analytical methods based on an assessment of actual-versus-predicted performance for use in design. An evaluation of current procedures should be performed with recommendations for the use of such procedures or the development of new procedures. These will include (but not be limited to) the following:

- a. Methods for calculating temperature distribution and resulting thermal stresses in a structural member, taking into account short-time transient effects as well as creep and stress relaxation.
- b. Analysis for calculating the effect of high temperature on the changing stress patterns of a component as a function of time, including the phenomenon of creep and stress relaxation.
- c. Analysis of structural members to predict changing deflection and distortion due to the effect of high temperature on the component as a function of creep, stress relaxation, and time.
- d. Effect on the above items by changes in material properties due to the following: radiation, sodium environment, fabrication, and instability of heavily loaded thin-wall members.

Task 5.1.C - Development of Inspection and Maintenance Techniques

The objective of this task is to develop techniques, procedures, and inspection frequencies for in-service inspection and to investigate the feasibility and procedures for corrective maintenance.

Until the long-term (30-year) environmental effects of sodium and radiation on material properties are well understood and supported by

statistical performance data, periodic in-service inspection of the vessel and related structural components will be required. Under this task, a criterion establishing the need and extent of inspection of various areas of the vessel will be developed based on an evaluation of that particular area and its susceptibility to failure, its mode of failure, failure effects on system safety, and what safety back-up systems are available. Any needed techniques, procedures, and frequencies will be developed from the assessment of the criteria.

A feasibility study on corrective maintenance techniques and procedures will be carried out. Such a study will investigate possible means of repair and suggest ways to accomplish such repairs in various areas.

The most promising vessel inspection components will be tested and the results evaluated. The selected reference inspection system then will be assembled and tested under simulated plant conditions to evaluate its performance.

Task 5.1.D - Closure and Seal Development

The objective of this task is to investigate and develop methods and associated hardware designs for vessel closures. This study includes the shield plug, static seals, seal faces, hold-down devices, and closure handling and guiding devices.

Because of its massive size, its geometry, and the demands it must meet, the shield plug presents several problems. The plug is required to support equipment with a minimum of deflection (such as control rods and drives). It must maintain long-term dimensional stability. Thermal distortion resulting from environmental conditions and internal heat generation must be minimal. It must exhibit rigidity, resist the effects of seismic forces, and contain internal energy releases.

The need exists for static seals that function efficiently in the operational environment for prolonged periods. Additionally, it would be advantageous to have seals with reusable qualities.

The remote operation of securing the shield plug and accurately removing and replacing is of concern due to the large size, the mass involved, and the close dimensional alignment required between operating mechanisms. The operational time involved to perform these operations is also of concern in this area.

The most promising designs of static seals, seal faces, and hold down devices will be screened and tested in an environment representative of its ultimate service. These tests will be performed in a chamber similar to the Westinghouse Glove Box Environmental Test Facility described in the Appendix. The tests will include a mock up of sodium and gas conditions as well as applied mechanical loads.

After these tests are completed, prototype tests will be performed in a closure and seal test facility mocking up reactor environmental conditions. The results of these tests then will be used to determine the mechanical design of the shield plug for the 1000 MWe LMFBR.

5.1.4 Relation to LMFBR Program Plan

The AEC approach to the liquid-metal reactor vessel as discussed in Task Area 3-9, "Reactor Vessels and Related Components" in Volume 3 of the LMFBR Program is in basic agreement with this plan.

The AEC recognizes the shortcoming of the present day codes and analytical methods, and accepts the need for the development of closure and inservice inspection methods.

5.1.5 Major Test Facility Requirements

The following facility is needed to accomplish the task.

Closure and Seal Test Facility (Task 5.1.D)

The facility shall be capable of testing the following:

- a. Operating pressure to 200 psi
- b. Operating temperatures to 1100°F
- c. Sodium-gas environment

- d. Prototype shield plug handling and guiding equipment
- e. Prototype shield plug and flanges to 10 foot diameter
- f. Various closure concepts
- g. Various seals different in material and configuration
- h. Gas cooling

5.1.6 Program Schedule

The time table for this development program is presented in Figure 5.1-2. Major milestone definitions, as indicated in scheduling figure, are given below.

Task 5.1.A - Development of Design Criteria

- 1. Initiate evaluation of task by reviewing the existing criteria
- 2. Input of material data from the materials study
- 3. Complete the final document

Task 5.1.B - Development of Analytical Techniques

- 1. Initiate a review of existing analytical techniques
- 2. Recommend findings from review of existing techniques
- 3. Complete the final analytical techniques

Task 5.1.C - Development of Inspection and Maintenance Techniques

- 1. Initiate the study of inspection and maintenance techniques
- 2. Define inspection criteria and areas to be inspected and initiate development of inspection equipment
- 3. Complete of development work for inspection equipment
- 4. Checkout test of inspection system completed and final analysis made of test

Task 5.1.D - Closure and Seal Development

- 1. Initiate design of closure test facilities
- 2. Completion of test facilities and initiate closure tests
- 3. Complete the closure test and initiate seal tests
- 4. Complete the seal test

Figure 5.1-2. Reactor Vessel Program Schedule

5.2 Main Transport Piping Development

5.2.1 Objective

To develop design, fabrication, and maintenance methods for large piping systems. The development effort must be sufficiently thorough so that the piping system will possess the exceptionally high reliability which is a requirement of all LMFBR systems.

5.2.2 Program Summary

The approach to piping system development is presented in Figure 5.2-1. To accomplish the objective, this program is divided into four tasks. The design method development task will be used to develop analytical design tools needed to guarantee the structural integrity of the piping systems under all possible operating modes. The support system development task will be used to determine the best methods for supporting piping system. The inspection and maintenance development program will be used to define main coolant requirements for the piping system. The fourth task will be used to evaluate and define piping and piping system fabrication techniques and procedures.

Since it is impractical to fabricate and test a large sodium piping system, the proving grounds for the above development work will be in the larger sodium test facilities (SPTF, SCTI, etc.) and Demonstration Plants.

5.2.3 Task Descriptions

Task 5.2.A - Design Method Development

The objective is to develop and/or modify existing design criteria, standards, and analytical methods to insure structural integrity during all transient and steady-state operating modes.

The design and analysis of piping systems to transport a multitude of fluids has been highly developed by chemical industries. However, the requirements of future LMFBR systems, to some extent, may make current analytical methods inadequate. The distinctive system features are:

1) a rather large diameter to wall thickness ratio (>65), 2) the

Figure 5.2-1. Model of Piping System Development Process

relatively high operating temperatures and plant characteristics which introduce the potential for severe transient conditions, 3) relatively large size pipes (48 inch diameter) and mass flow, and 4) the requirement of an exceptionally high reliability. The product of this analytical study will be design, criteria standards, and analysis methods which specifically account for the above physical and operational requirements.

Task 5.2.B - Support System Development

The objective is to evaluate and/or develop pipe support hardware, support system design, and operation of the system during loading and unloading operations, and during transient and steady state plant operation.

The design of a proper support system cannot be over-emphasized with regards to maintaining the structural integrity of the piping network. The system must adequately support the load of both empty and loaded pipes (weight difference is as much as 700 pounds per foot of pipe) without suppressing the systems' expansion characteristics. This study will evaluate commercially available hangers, including a) solid hangers, b) constant support hangers, and c) variable spring hangers. Recommendations will be made regarding the placement, operation, and maintenance for each of these units. Where inadequacies exist, alternate support methods and hardware will be devised.

This study will be coordinated closely with Task 5.2.A, since the overall design and analysis of the pipe system will largely depend on the support hardware available.

Task 5.2.C - Inspection/Maintenance Program Development

The objective is to define inspection, surveillance, and corrective maintenance programs for the main coolant piping system. The necessity for inspections, inspection requirements, methods, and frequencies will be resolved. Corrective maintenance problems, such as access requirements, plant availability, and high background radiation will be considered.

Due to the obvious access difficulties, stringent environmental conditions, and the permanence of the piping system, development of inspection techniques and equipment which will produce reliable assurance of the continued structural integrity of the system is necessary. Development of repair and test equipment and techniques is difficult due to the many and varied failure modes which could be conceived. However, corrective maintenance concepts must be developed. Consideration of these concepts early in the design stage will aid the maintenance procedures being developed, should they be necessary during plant lifetime.

There is no immediate need to develop inspection and maintenance programs and equipment. However, concepts and criteria should be established early to allow for consideration in the plant preliminary design stage.

Task 5.2.D - Fabrication Development

The objective is to 1) specify fabrication procedures and techniques, 2) establish a "sequence of assembly" which will yield the minimum cost, 3) compare "shop weld" quality to "field weld" quality, 4) define quality control practices, and 5) develop any related special equipment.

The piping system is a complex of many similar individual parts joined together by a common process. Once a reliable and simple process is perfected for welding the joints, the procedure can be repeated to yield a reliable system at a minimum cost.

The piping system may be assembled either in the shop or in the field. Field assembly tends to be costly and may result in a lower quality product. For this reason, "sequence of assembly" should be devised which will permit a maximum amount of work to be performed in the shop under maximum control. It is realized that some field work will be required; therefore, special development should be planned in this area to produce the highest possible quality product.

In addition to the above, other related fabrication topics will be evaluated and/or defined. These include; 1) quality control, 2) inspection procedures, 3) tolerances, and 4) training and orientation

programs for workmen.

Fabrication and assembly of piping systems for large sodium test facilities and the FFTF and Demonstration Plants will be followed and evaluated as part of this task.

5.2.4 Relation to LMFBFR Program Plan

The AEC plan for piping system development as described in Task Area 3-8, "Coolant Transport and Storage Systems" in Volume 3 of the LMFBFR Program Plan, is considered complete, with one minor exception. The problem of piping support for large diameter pipe is identified, but no specific effort is allocated in the tasks to the design and evaluation of piping supports. This program could be included as part of Task 5.2.D, "Development of Designs for Coolant-Transport Systems".

Westinghouse agrees with the AEC statement that in the past, piping system design has been handled by the architect-engineer in accord with handbook solutions, and that such practices are not suitable for LMFBFR systems.

5.2.5 Major Test Facility Requirements

No major test facilities are required directly. All verification of method and hardware will be part of the work performed in the design, construction, and operation of large sodium test facilities needed for other programs and the FFTF and Demonstration Plants.

5.2.6 Program Schedule

The timetable for this development program is presented in Figure 5.2-2. Major milestone definitions, as indicated in the scheduling figure, are presented below.

Task 5.2.A - Design Method Development

1. Initiate evaluation of design methods
2. Design criteria formulated and analytical methods available.

Task 5.2.B - Support System Development

1. Initiate evaluation of existing support hardware
2. Support hardware and application technology available for large diameter piping systems.

Task 5.2.C - Inspection and Maintenance Development

1. Initiate study of inspection and maintenance requirements
2. Maintenance requirements defined, and associated methods and equipment conceived.

Task 5.2.D - Fabrication Development

1. Initiate study of piping fabrication
2. Basic fabrication programs complete
3. Evaluation of piping fabrication methods used for SPTF (approximately 30 inch diameter pipe)
4. Evaluation of piping fabrication methods used for FFTF (approximately 14 inch diameter pipe)
5. Evaluation of piping fabrication methods used for Demonstration Plant (approximately 24 inch diameter pipe)

Figure 5.2-2. Main Transport Piping Development Program Schedule

5.3 Check Valve Development

5.3.1 Objective

To develop sodium valve technology so that the design and fabrication of a 48 inch diameter check valve can be accomplished successfully. Success is dependent on: 1) the unit's overall reliability, 2) associated costs, and 3) operational performance.

5.3.2 Program Summary

The plan for achieving the objective is shown schematically in Figure 5.3-1. Separate studies will investigate the major problem areas associated with check valves (distortion, oxide deposition and self welding, and sodium hammer). The findings and recommendations will be collected and utilized in the design and fabrication of developmental and production (FFTF and Demonstration Plant) check valves. The reference valve will evolve finally from the prototype programs, which will progressively increase the size of valves being built and tested.

The development of check valves will be correlated closely with sodium pump development, the system pump being the primary factor in determining the valve's functional and operational characteristics.

5.3.3 Task Descriptions

Task 5.3.A - Concept Evaluation and Selection

The objective is to review and evaluate basic check valve concepts to determine the type best suited for specific applications in sodium systems. The objective will be reached primarily through the preparation of numerous conceptual valve designs for a wide range of operating parameters.

Currently, large check valves (24 inch) are readily available to the chemical and power industries. The basic designs employed in those industries (swing type, center guided, etc.) are directly applicable for sodium service. The unique characteristics of a sodium system (high temperature, relatively severe transient) however, will demand

Figure 5.3-1. Model of Check Valve Development Process

modifications in the detail design, and in the general construction of conventional units in order to produce an acceptable piece of "sodium hardware." In addition, the size valve required and the particular placement in the system will dictate the most suitable valve type. This task will resolve the above problems, and the results will be utilized not only in the reference plant design, but also in planned Demonstration Plants and large sodium test facilities which will precede the reference plant.

Task 5.3.B - Thermal Studies

The objective is to investigate the effects of the thermal environment on critical valve components. This study will be analytical in nature, with verification of results coming from prototype tests. Failure to accurately identify and account for resulting thermal stress and/or deformations could lead to premature failures or valve malfunctions.

The characteristic, high operating temperatures and extremely rapid thermal transients combine with the low thermal resistance of sodium films to render check valves vulnerable to severe thermal shocking or distortion. Such conditions may interfere with valve actuation and may destroy tight shutoff capability. Since the initiating causes of this problem cannot be eliminated, efforts will be made to minimize their effects by proper design. Therefore, specific design details and special features will be developed to absorb transient effects and thereby, prevent valve damage.

Task 5.3.C - Oxide Deposition and Self-Welding Studies

The objective is to study the effects of sodium oxide deposition and/or self-welding on valve operation. Methods for preventing these problems in critical areas, and/or design features which will allow proper operation should these problems exist, will be developed.

Experience indicates that deposition of sodium oxide on bearing or guide surfaces or self-welding can alter the response characteristics. The worst case occurs when the check valve sticks in an open position. The study will investigate the mechanisms by which oxide deposition and self-welding

occur and develop methods for annulling and/or minimizing the resulting effects. Experimental investigations, along with analytical efforts, will be performed at a laboratory level.

Task 5.3.D - Closure Control Studies

The objective is to develop "add on" devices for existing check valve designs, and/or modify the entire valve design so that the operational rate of flow shutoff will not produce "sodium hammer" in the closed system.

"Sodium hammer" will result from a sudden flow stoppage in a closed loop; poorly designed check valves can instigate "sodium hammer." If the travel of the check valve disc lags the flow coastdown, the flow will reverse before the disc is seated. If there is no provision for damping the disc velocity as it approaches the seat, the reversed flow will slam the disc into the seat, resulting in "sodium hammer". Design features, such as dashpots, can be used to control the rate of flow shutoff. Preliminary experimental investigations can be performed in a simple water facility. Final proof tests for closure performance will be included in the test program for prototype valves (Task 5.3.E).

Task 5.3.E - Prototype Valves

The objective is to fabricate and test complete check valve units to verify the performance of the entire package in an operational environment. The early phases of this task will concentrate on tests of small (less than 10 inches) valve units in a water facility. The valve types are determined by Task 5.3.A efforts, and the main area of investigation will correspond to work of Task 5.3.D. Developmental valve prototypes for sodium service will be built and tested parallel with sodium prototypes in the sodium pump test facility (SPTF). This will include at least a 12,000 gpm unit (12-14 inches), and a 60,000 gpm unit (28-30 inches). Success with the above units will make possible the fabrication of the reference unit with an acceptable risk margin; only standard post-production tests will be necessary.

In addition to the above prototypes, operational analysis of check valves in the FFTF and Demonstration Plant will be available.

5.3.4 Relation to LMFBR Program Plan

The AEC approach to check valve development as described in Task 3-8.5, "Development of Valves," in Volume 3 of the LMFBR Program Plan is well defined. The method of solution appears reasonable and complete. The AEC philosophy of "duplication of valve design and development effort by more than one organization is desirable -- to develop industrial competence and to provide alternative components for use by system designers" is especially logical.

The one minor criticism is the failure of the AEC to point out the relationship between sodium pump development and check valve development. It is stated that prototype testing will be centralized in a LMEC facility; however, it does not specifically mention which one. Westinghouse feels that the sodium pump test facility should be utilized for check valve prototype tests. All environmental parameters (temperature, flow, etc.) will be duplicated in the SPTF, including the effect of the flow variance produced by the pump on the check valve operation.

5.3.5 Major Test Facility Requirements

The functional requirements, associated tasks, and descriptive titles of the major test facilities are listed below.

Hydraulic Test Facility (Tasks 5.3.D and 5.3.E)

Provide 15,000 gpm of water. This is the same facility used in the Task 5.4.C section of the sodium pump development program.

High Temperature Sodium Loop (Task 5.3.E)

The sodium pump test facility (SPTF) will fulfill this need. Functional specifications for this facility are presented in the technical program for pump development.

5.3.6 Program Schedule

The timetable for this development program is shown in Figure 5.3-2. Major milestone definitions, as indicated in the scheduling figure, are presented below.

Task 5.3.A - Concept Evaluation and Selection

1. Initiate concept evaluation studies.
2. Recommend the valve types which will receive further study and development.

Task 5.3.B - Thermal Studies

1. Initiate analytical investigations of thermal problems.
2. Complete analytical investigations of thermal problems. All related experimental and proof testing work will be included in prototype testing.

Task 5.3.C - Oxide Deposition and Self Welding Studies

1. Initiate evaluation of oxide deposition and self welding phenomena.
2. Complete study in sufficient detail to fully understand deposition problems and formulate corrective actions for future developmental valves. Experimental verification will be part of prototype tests.

Task 5.3.D - Closure Control Studies

1. Initiate valve closure control studies.
2. Complete study with recommendations on valve designs which will yield acceptable closure responses. Verification tests are part of prototype programs.

Task 5.3.E - Prototype Check Valves

1. Initiate fabrication of test check valves.
2. Complete "short term" tests for the 12-14 inch (12,000 gpm) check valve. Coincident with 12,000 gpm pump tests.
3. Check valve in the FFTF operation.

4. Complete long term tests on 28-30 inch (60,000 gpm) check valve. Coincident with 60,000 gpm pump tests. A success here will permit direct extrapolation to a 48 inch unit.
5. Demonstration Plant check valves (24 inch) operational.

Figure 5.3-2. Check Valve Development Program Schedule

5.4 Large Sodium Pump Development

5.4.1 Objective

To develop sodium pump technology to a level where the design and fabrication of units which meet the functional requirements of the reference 1000 MWe LMFBF plant can be accomplished without undue technical and economic risks.

5.4.2 Program Summary

The approach to pump development is depicted schematically in Figure 5.4-1. The program is divided into seven major work tasks. The first six involve analytical and experimental investigations which provide the technological foundation for prototype testing, which is the seventh task. Prototype testing includes out-of-pile tests on developmental pumps (12,000 and 60,000 gpm units), as well as operational analysis of FFTF and Demonstration Plant units. The seven tasks described in the next section are listed below.

- Shaft Seal Development
- Bearing Development
- Hydraulic Development
- Thermal Studies
- Dynamic and Vibration Studies
- Mechanical Studies
- Prototype Testing

5.4.3 Task Descriptions

Task 5.4.A - Shaft Seal Development

The objective is to develop a gas shaft seal for large shaft diameters (8-18 inches) and shaft speeds up to 1200 rpm. Desirable seal characteristics are: a) a high degree of reliability, b) a minimum life expectancy of one year, c) a very low power consumption, and d) ease in manufacture, installation, and maintenance.

Failure to achieve all of the above will affect overall pump reliability, add to fabrication and maintenance costs, and increase pump operating costs.

Figure 5.4-1. Model of Large Sodium Pump Development Process

The function of the shaft seal is to maintain a boundary along the pump shaft between the argon-sodium vapor cover gas and the pump vault atmosphere. Existing operational shaft seal designs (mostly of the "oil" type) have an average operating life of about six months. The seals require precision assembly and installation, are susceptible to thermal distortion, and often have a higher leakage rate than expected. To add to this discouraging picture, none of the seals are applicable to the shaft diameters of interest; many of the shaft seal problems are magnified as the size increases.

The experimental and analytical studies will investigate: a) material performance, b) leakage rates (effects of temperature and pressure), c) effects and type of failures, d) wear and expected life data, e) prooftesting of seal designs in the Shaft Seal Test Facility, and f) evaluation and prooftesting of seals in operating pumps.

Task 5.4.B - Bearing Development

The objective is to develop radial bearings. The maximum journal diameter considered is 26 inches. Primary bearing development goals are: a) high reliability, b) long life, c) ease in maintenance, d) low power consumption, and e) ease of manufacture. Failure to achieve all of the above will affect overall pump reliability, add to fabrication and maintenance costs, and increase pump operating costs.

Currently, three bearing types appear most promising for future applications. They are: 1) self-aligning hydrostatic, 2) non self-aligning hydrostatic, and 3) pivoted pad hydrodynamic. This program will pursue development of all three bearing types since optimum bearing types vary with particular applications (e.g., pump size, pump type, etc.).

Aside from obtaining the analytical tools required for the design of bearings, a program for the measurement of actual pump bearing loads is desirable. Test data will be obtained that will enable a bearing to be designed for any given conditions, and experimental determinations of the actual bearing load will permit the designer to size a bearing more accurately for a given application. In addition, the materials and method of fabrication must be evaluated for large diameter bearings.

This program will include: a) determining bearing loads (this will be done with hydraulic models in Task 5.4.C), b) bearing material evaluation and associated fabrication techniques, c) proof testing of bearing designs in the Bearing Test Facility, and d) continued evaluation of bearing performance in prototype and operational pumps.

Task 5.4.C - Hydraulic Development

The hydraulic development portion of this program is divided into two distinct sections: a) the impeller area hydraulics and, b) the behavior of the free surface sodium in the pump tank. These two sections are combined into one task because experimental work for both could be performed simultaneously. All tests will be conducted in a water facility with representative models (minimum scale is one-tenth pump size). Impeller area hydraulic tests will be required for each pump concept considered (not dependent on pump size). Pump tank hydraulic studies can be performed with the above models, or in separate, simplified models.

- a. Impeller Area Hydraulics - The objective is to develop and optimize pump hydraulics (approach channels, impeller inlet areas, impeller blade design, guide vanes, volutes, etc.) to obtain maximum efficiency with minimum radial thrust and minimum required NPSH. Additionally, data will guarantee that the actual performance will match that required. Model testing will be carried out in order to verify hydraulic performance and radial thrust values for bearing development.
- b. Pump Tank Hydraulics - The objective is to study the behavior of the sodium level in the pump tank and the resulting effects on pump operation. Failure to control the sodium level could cause pump tank flooding with subsequent seal damage, or loss of pump tank level and subsequent gas entrainment. Experimental investigations will be conducted to determine; a) sodium level control methods and effectiveness, b) effects of sodium level variance, and c) potential for gas entrainment.

Task 5.4.D - Thermal Studies

The objective is to investigate the effects of the thermal environment on critical pump components. The study will be analytical in nature, with verification of results coming from prototype tests. Failure to accurately identify and account for resulting thermal stresses and/or deformations could lead to premature failures or binding of rotating and/or removable elements.

The use of sodium in heat transfer systems results in high thermal stresses in structural materials due to the low thermal resistance of sodium films. For all developmental and production pumps, a complete thermal stress analysis must be performed during the preliminary design of all subassemblies. Areas where close clearance must be maintained will be analyzed to determine the effects of thermal expansion.

Task 5.4.E - Dynamic and Vibration Studies

The objective is to conduct analytical vibration studies on the pump shaft assemblies. The assembly includes the impeller, shaft, and motor rotor. Any level short of a thorough understanding of this subject could lead to a complete pump failure early in its operational life. Specific topics for investigation are: a) effects of torsional and longitudinal frequencies, b) transmission of shaft vibrations to pump body and mountings, c) damping methods, and d) relationships between vibration frequencies, shaft lengths, masses, speeds, and geometries.

Task 5.4.F - Mechanical Studies

The objective is to develop and test static seals for slip fits to minimize sodium leakage from the pump discharge casing into both the pump intake and the pump tank areas.

Aside from the structural design, consideration must be given to assembly, disassembly, and maintenance of the unit. It will be necessary to assure that the pump internals and shaft assembly can be inserted into or removed from the fixed pump tank assembly. The ideal design would be one which would make it possible to withdraw the entire inside assembly including the impeller, diffuser, thermal barrier, the lower bearing, shaft and baffling from the pump tank.

Some type of static seal will be required to minimize or prevent leakage between the pump tank and suction and discharge slip fit areas. Since there will be relative motion between these parts, due to thermal expansion, the seal will have to be a flexible or sliding seal.

Task 5.4.G - Prototype Testing

The objective is to verify, reinforce, and combine the results of the six previous tasks by means of progressive fabrication, testing, and operational analysis of increasingly larger pumps. Test data will provide feedback to analytical studies and subcomponent tests to guide further basic development programs. Prototype testing includes sodium pump test facility (SPTF) tests, and FFTF and Demonstration Plant operation.

5.4.4 Relation to LMFBR Program Plan

The AEC approach to liquid-metal pump development as described in Task Area 3-11, "Liquid Metal Pumps," in Volume 3 of the LMFBR Program, generally is in good agreement with the plan presented here.

The AEC plan accepts the centrifugal pump as a pump type which displays the most potential for meeting the demands of future LMFBR's. The Westinghouse approach to development and recognition of specific problem areas coincides with the basic approach of the AEC.

The one point in the AEC plan which does not, on the surface, appear to adequately concur with Westinghouse needs is "The development of pumps larger than 60,000 gpm is not now planned." A closer examination of the problem reveals that the proposed 150,000 gpm pump represents an extrapolation of only about 25 percent when critical pump dimensions are compared (e.g., maximum tank diameter for 60,000 gpm unit is approximately 110 inches; that for the 150,000 gpm unit it is approximately 145 inches.) This stems from the difference in functional requirements for each pump. Approximate values are compared on the following page:

Flow rate, gpm	150,000	60,000
Head, feet of Na	300	350
NPSH available, feet of Na	100	40
Speed, rpm	900	400

The only requirement for the reference pump which contributes to an increase in size is the flow capacity; all other requirements and characteristics are instrumental in minimizing the pump size. In summary, pumps cannot be compared solely on the basis of flow capacity.

5.4.5 Major Test Facility Requirements

The functional requirements, associated tasks, and descriptive titles of the major test facilities are listed below:

Shaft Seal Test Facility (Task 5.4.A)

- a. Accept shaft seals up to 18 inches in diameter.
- b. Provide variable shaft rotation up to 1200 rpm.
- c. Vary shaft alignment and eccentricity in a controlled fashion.
- d. Emit sodium vapor into gas lubricant.
- e. Provide a 100 psi pressure differential across seal.
- f. Provide a thermal environment of 1200°F.

Bearing Test Facility (Task 5.4.B)

- a. Accept bearings up to 26 inches in diameter of journal.
- b. Provide variable shaft rotation up to 1200 rpm.
- c. Vary alignment of shaft and load on bearing in an orderly fashion.
- d. Provide thermal environment of 1200°F.
- e. Produce 35°F/second thermal transients for a period of 6 seconds.

Hydraulic Test Facility (Task 5.4.C)

- a. Provide 15,000 gpm of water.

Sodium Pump Test Facility (Task 5.4.G)

- a. Accept pumps up to 120,000 gpm.
- b. Provide pumping resistances up to 450 feet of Na.

- c. Supply pumps with NPSH up to 200 feet of Na.
- d. Provide sodium thermal environment of up to 1200°F.
- e. Produce 35°F/second thermal transients for a period of 6 seconds.

5.4.6 Program Schedule

The schedule for this development program is presented in Figure 5.4-2. Major milestone definitions, as indicated in the scheduling figure, are presented below.

At present, Westinghouse is performing a conceptual pump design program, "Large Sodium Pump Study," AT (30-1)3762, which is funded by the AEC. The seven tasks which are described have been initiated and are now in the conceptual design phase.

Task 5.4.A - Shaft Seal Development

- 1. Complete design, fabrication and verification tests for seal on 12,000 gpm prototype.
- 2. Complete design, fabrication and verification tests for seal on 60,000 gpm prototype.
- 3. Complete design, fabrication and pre-production tests on seal for 150,000 gpm reference pump.

Task 5.4.B - Bearing Development

- 1. Complete design, fabrication, and verification tests for bearing on 12,000 gpm prototype.
- 2. Complete design, fabrication, and verification tests for bearing on 60,000 gpm prototype.
- 3. Complete design, fabrication, and pre-production tests for bearing on 150,000 gpm reference pump.

Task 5.4.C - Hydraulic Development

- 1. Complete hydraulic investigations for 12,000 gpm prototype pump.
- 2. Complete hydraulic investigations for 60,000 gpm prototype. If 60,000 gpm pump and 150,000 gpm reference pump are of the same basic design (this a valid assumption), these data will be directly applicable to 150,000 gpm design effort.

Task 5.4.D - Thermal Studies

1. Complete analytical investigations of thermal problems. All related experimental and proof testing work will be included in prototype testing.

Task 5.4.E - Dynamic and Vibration Studies

1. Complete analytical investigations of dynamic and vibration problems. All related experimental and proof testing work will be included in prototype testing.

Task 5.4.F - Mechanical Studies

1. Complete study in sufficient detail and scope so that it will be adequate for entire program. Additional related work will continue in individual prototype programs.

Task 5.4.G - Prototype Pumps

1. Complete proof testing 12,000 gpm prototype. No long term tests will be conducted on this unit.
2. FFTF pumps operational.
3. Complete entire test program for 60,000 gpm prototype. The functional requirements of this unit (350 feet head, 40 feet NPSH available, low rpm) are different from the reference 150,000 gpm unit (300 feet head, approximately 100 feet NPSH available, high speed). However, the net effects result in an extrapolation of only about 25 percent in physical dimensions above the 60,000 gpm prototype. Therefore, if the 60,000 gpm program is successful, it may be feasible to proceed directly to the 150,000 gpm reference pump design.
4. Demonstration Plant pumps (approximately 40,000 gpm) operational.

Figure 5.4-2. Large Sodium Pump Development Program Schedule

5.5 Sodium-to-Sodium Heat Exchanger Development

5.5.1 Objective

To develop the sound technological base necessary to permit the safe, reliable design, fabrication, and operation of sodium-to-sodium heat exchangers within the criteria established for the reference 1000 MWe LMFBR plant.

5.5.2 Program Summary

The approach to intermediate heat exchanger development is shown in Figure 5.5-1. The program is divided into six major tasks each one identified as requiring research and development efforts in order to ensure safe, reliable, and predictable unit performance throughout the duration of its design life. The first three tasks deal primarily with the analytical and experimental investigations which provide the technological foundation, such as design, fabrication and maintenance, for model and prototype testing.

Prototype testing includes models up to one MWt capacity and one 30 MWt capacity prototype. The intermediate milestone for IHX development is identified as the 340 MWt unit for a Demonstration Plant.

Since the reference IHX has many areas requiring development similar to those of a steam generator, (Program 5.6) this intermediate heat exchanger program is viewed as a program complementing steam generator development activities.

5.5.3 Task Descriptions

Task 5.5.A - Thermal and Hydraulic Development

The objective of this task is to provide thermal-hydraulic design information for the intermediate heat exchanger. The proposed activities complement the results of similar steam generator development task (Task 5.6.A).

Subtask 5.5.A.a - Sodium Distribution Simulation Test (Shell and Tubes)

The objective of this subtask is to investigate and evaluate the fluid-flow characteristics over the reference IHX's shellside geometry and in

Figure 5.5-1. Model of Na-to-Na Heat Exchanger Development Process

the tubes to achieve optimum flow distribution. This task also will be used to evaluate the design and arrangement of nozzles, internals, etc., and their resulting effects on flow distribution. Special emphasis will be placed on proper flow characteristics to: a) ensure good heat transfer qualities, and b) minimize flow-induced tube vibration.

Initial simulation tests will be performed using small model water tests to verify analytical techniques and as a method of screening alternate concepts. Evaluation of sodium distribution for each succeeding sodium-to-sodium model test up through the Demonstration Plant IHX will be conducted.

Subtask 5.5.A.b - Thermal Baffle Development

The objectives of this subtask are to a) evaluate the behavior of critical IHX parts under thermal transients and the influence of thermal baffles on this behavior, b) design suitable protective baffles where necessary, and c) verify the performance of critical IHX parts by subjecting them to thermal shock tests. This subtask is similar to that proposed for the steam generator (Subtask 5.6.A.c) and will utilize results of that task as a basis for work.

Of prime interest in the IHX with regard to transient conditions are: a) tube-to-tubesheet welds, b) shell-to-tubesheet welds, and c) main coolant nozzles. The transients of most concern are loss of sodium on either the shell or tube side. The proposed program would simulate both of these incidents and subject the models to a greater number of cycles than the part would ever experience in service.

Subcomponent thermal shock tests, initially, will be performed to evaluate the necessity of incorporating thermal baffles in critical areas. Larger model tests (1 MWt and 30 MWt), FFTF and Demonstration Plant sodium-to-sodium heat exchangers will be evaluated to determine effectiveness of baffles where incorporated. The results of these evaluations will be used as design input for the 1000 MWe LMFBR plant.

Task 5.5.B - Development of Fabrication and Quality Control Technology

The objective of this task is to develop a fabrication and quality control technology consistent with the safety and reliability demands established by design criteria. Fabrication and quality control procedures will be updated and re-evaluated continually as larger models, up through the Demonstration Plant IHX, are built and operated.

Subtask 5.5.B.a - Fabrication/Manufacturing Process Development

The objective of this subtask is to develop and verify the fabrication techniques which will be applied to the proposed IHX unit. Emphasis will be on efficient manufacturing methods which will yield a high quality product. The program will include: a) specifying manufacturing tolerances, b) design of assembly jigs and fixtures, c) defining quality control practices, d) evolution of training and orientation programs for workmen.

To ensure the best product possible, representative model sections will be constructed in order to a) develop optimum fabrication procedures, and b) verify that acceptable inspection of the work can be performed to determine product quality. If these two goals are achieved, the probability of a component failure will be minimized.

Subtask 5.5.B.b - Quality Control Development

The objective of this subtask is to develop quality control and inspection techniques and procedures consistent with the safety and reliability requirements established by the design criteria. The state-of-the-art will be reviewed and industrial practices will be assessed and revised as necessary. The goal of fabrication quality control is to detect, as early as possible, any substandard conditions that may arise. A coordinated activity, incorporating provisions for the periodic feedback of test data to the overall reliability assurance group, will be initiated to ensure proper dissemination and evaluation of latest information.

Subtask 5.5.B.c - Tube-to-Tubesheet Joint Development

The objective of this subtask is to a) devise, b) develop, c) test, and

d) evaluate tube-to-tubesheet attachment methods. This program is similar to that for tube-to-tubesheet attachment in the steam generator (Task 5.6.B.b). The differences between tasks are: 1) the IHX uses stainless steel tubes and tubesheets, while the steam generator joins Incoloy tubes to Inconel-clad stainless sheets, and 2) the development program for IHX tube attachments will lag the steam generator task in order to benefit from its experience.

From a reliability standpoint, past experience demonstrates that the tube-to-tubesheet attachment in shell and tube heat exchangers is, perhaps, the most critical area. The delicate joint formed by the relatively thin tube walls and thicker tubesheet is readily susceptible to thermal shock damage, vibrational damage, and possible corrosion. Several attachment methods have been proposed, and many more possibilities exist. This study should determine a) which is the best attachment method and b) what type of performance can be expected from it.

Task 5.5.C - Maintenance Development

The objective of this task is to a) define probable maintenance requirements, and b) develop and prove techniques, equipment, and procedures for inspection, repair, and retesting.

Even though the IHX is the end product of a comprehensive design program where operational reliability has been emphasized, the unit is not maintenance free, nor would a unit malfunction be beyond reality. This fact, combined with the heavy penalty and inconvenience of plant "down time" after attaining an operational status, makes necessary that prior to installation, maintenance procedures are developed and proven.

The problem involves:

- a. Adequate inspection techniques to prevent total failures.
- b. Defining the degree of deterioration before repairs must be made.
- c. Disassembly of the unit to the extent where the repair can be made.
- d. Testing the repaired area to ensure "required" performance.
- e. Reassembly

If the current development of equipment, techniques and procedures can expedite any of the above areas and reduce the chance of a reoccurrence of such an incident, then a worthwhile economic gain in plant utilization may be achieved.

Along with procedures, maintenance equipment components will be fabricated and will be tested to develop the required product. A total maintenance system will then be built for use with the 30 MWt model. Results of these tests, along with FFTF and Demonstration Plant maintenance, and equipment evaluations will be used for the 1000 MWe LMFBR reference system design.

Task 5.5.D - Follow-On Program Definition and Prototype Planning

The objectives of this task are 1) to identify and scope the additional research and development necessary to achieve safe, reliable, and economic sodium-to-sodium heat exchangers, and 2) to plan for the design, fabrication, and testing of heat exchanger models.

Subtask 5.5.D.a - Program Definition

The objective of this subtask is to identify and scope additional research and development activities. The basis for recommended research and development programs will be a review and evaluation of the results of earlier tasks and other AEC-sponsored component development programs. A number of the programs are expected to be a continuation or expansion of the Technical Support Programs, Tasks 5.5.A, 5.5.B and 5.5.C.

Subtask 5.5.D.b - Prototype Planning

The objective of this subtask is to plan for the design, fabrication, and testing of heat exchanger models. The test models will be of a size adequate for "proof of design" of the large components. The program to test these models will be scoped, assuming AEC facilities will be available for testing. The test program plan will cover both steady-state operation and transients testing for the system-imposed design limits.

Task 5.5.E - Model and Prototype Testing

The objectives of this task are to design, fabricate, test, and evaluate

model sodium-to-sodium heat exchangers. Conceptual design emerging from previous tasks will be detail designed, according to functional and design criteria of the reference plant. Fabrication of models will utilize the latest specifications and techniques developed for welding and quality control.

Subtask 5.5.E.a - Model Tests (<1 MWt)

Models with capacities of less than 1 MWt will be designed and tested as preliminary proof-of-principles tests for various designs. A model between 1-2 MWt capacity will be designed and tested to incorporate many of the features of the 1300 MWt and 30 MWt designs. This model will serve to evaluate the chosen design on a small and relatively inexpensive scale before the design and fabrication of the 30 MWt unit is complete. Information gained in these tests will be factored into the design and used to specify the IHX test program in the SCTI. Design and materials data derived from the test also will be used in a revised 1300 MWt design.

Subtask 5.5.E.b - Prototype Testing (30 MWt)

A 30 MWt prototype sodium-to-sodium heat exchanger shall be designed so that its manufacture and planned test program will evaluate and demonstrate the principles of design, performance, and reliability required in a 1300 MWt unit. Various portions of the 30 MWt prototype will be full-size duplicates, while other features will be scaled. Both techniques will provide valid design criteria for full-scale units.

The 30 MWt prototype also must incorporate special test features so that the performance evaluation data may be gathered. Accordingly, the unit will be designed with integral temperature, pressure, flow, liquid-level, and strain-gage instrumentation.

Performance data applicable to the final design of a full-scale unit will include the following:

- a. Overall heat transfer performance at part and full load.
- b. Fluid flow characteristics at part and full load.
- c. Transient response behavior including flow stability.
- d. Thermal and mechanical behavior of component parts under cyclic load.

Subtask 5.5.E.c - 340 MWt IHX - Demonstration Plant

The intermediate goal for the development of sodium-to-sodium heat exchangers is to provide a reliably operating unit for the 300 MWe Demonstration Plant.

Task 5.5.F - Reliability Analysis

A principal objective of this task and of the total program is to ensure that the developed heat transport system is reliable and safe. Reliability analyses will be factored in all phases of IHX design, fabrication, and development.

Activities to meet the task objective will involve the following:

- a. Coordination of Reliability Efforts
- b. Incorporation of Reliability in Design Reviews
- c. Specific Reliability Reviews
- d. Design of Test and Analysis of Results
- e. Fabrication Reliability

5.5.4 Relationship to LMFBR Program

The AEC approach to the development of intermediate heat exchangers as described in Task Area 3-12, "Heat Exchangers" in Volume 3 of the LMFBR Program is in general agreement with the plan presented herein.

5.5.5 Major Test Facility Requirements

Because many of the task activities for steam generator development are applicable to intermediate heat exchanger development, only one additional major facility will be required to fulfill IHX development.

Sodium-to-Sodium Test Facility (Subtask 5.5.E.b)

The purpose of this facility is to provide for testing IHX's with capacities to 30 MWt. The facility should be designed to operate at steady-state conditions and also be provided with transient test capabilities. It is quite possible that this facility may be the result of modification of steam generator test facilities. If this can be achieved, however,

careful attention and coordination will be required to secure the proper amount of loop availability in order to satisfactorily proof-test the IHX.

5.5.6 Program Schedule

The intermediate heat exchanger development schedule is given in Figure 5.5-2. The milestones shown in the figure are described below.

Task 5.5.A - Thermal and Hydraulic Development

1. Initiate Thermal and Hydraulic evaluation studies
2. Complete Model Flow Tests
3. Complete analysis of Model Test Data

Task 5.5.B - Development of Fabrication and Quality Control Technology

1. Initiate task with a review of state-of-the-art technology
2. Complete development of the tube/tubesheet attachment
3. Complete Fabrication and Quality Control Specifications

Task 5.5.C - Maintenance Development

1. Initiate study of state-of-the-art technology
2. Issue Design and Procedural Specifications for the Maintenance System

Task 5.5.D - Follow-On Program Definition and Prototype Planning

1. Initiate Program Definition Task defining Heat Exchanger development program
2. Identification of Follow-On Tasks

Task 5.5.E - Model and Prototype Testing

1. Initiate 1 MW Model Test Program
2. Complete 1 MW Model Test
3. Complete 30 MW Prototype Test

Task 5.5.F - Reliability Analysis

1. Initiate reliability requirement studies
2. Establish preliminary procedures and methods for reliability analysis

Figure 5.5-2. Intermediate Heat Exchanger Development Program Schedule

5.6 Steam Generator

5.6.1 Objective

To develop the sound technological base necessary to permit the safe, reliable design, fabrication, and operation of sodium-heated steam generators within the criteria established for the 1000 MWe LMFBR reference plant.

5.6.2 Program Summary

The approach to steam generator development is shown schematically in Figure 5.6-1. The program is divided into six major work tasks. The first three deal primarily with the analytical and experimental investigations which provide the technological foundation for model and prototype testing. The remaining three tasks provide for the identification of follow-on research and development programs, reliability analysis, and for the planning and testing of prototypes. Prototype testing includes models up to 1 MWt capacity and one 30 MWt capacity prototype to be tested at the AEC-SCTI facility. The intermediate milestone toward the 1300 MWt target steam generator (1000 MWe Plant) is identified as the 340 MWt unit for a Demonstration Plant.

This program will develop sufficient information to permit accurate thermal, hydraulic, and mechanical analyses required for the safe, reliable design of a steam generator. The main information being sought is 1) the verification, refinement, and/or development of techniques of analysis to determine thermal performance, flow stability, flow distribution, tube vibration, and transient-temperature distributions in steam generators, 2) the establishment of analytical techniques (based on experimental data) for predicting the consequences of a sodium water reaction, and 3) verification, refinement, and/or development of techniques to ensure the structural and mechanical integrity of a sodium-water heat exchanger.

This program will include, under the respective task activity, the evaluation of analytical and experimental techniques utilized as the methods of analysis for all current steam generators designs, as well as those steam generator designs under development or to be developed. Recommendations for preferred

Figure 5.6-1. Model of Steam Generator Development Process

analytical and experimental techniques of steam generator analysis resulting from these activities will enable the achievement of the program objective of developing a 1300 MWt steam generator.

5.6.3 Task Descriptions

Task 5.6.A - Thermal and Hydraulic Development

The objective of this task is to provide thermal-hydraulic design information for heat transfer components. The activities proposed for this phase of the development program are designed to minimize design uncertainties and provide the experimental data necessary to design economic components.

Subtask 5.6.A.a - Shell Side Sodium Distribution Simulation Test

The objective of this subtask is to investigate fluid-flow characteristics over the reference steam generator's shell side geometry. Various other shell side geometries will also be investigated in an attempt to obtain an optimum geometry. Uniform sodium flow distribution on the shell side, without channeling or local recirculation, is necessary for efficient performance. Small-scale flow distribution models will be made to represent shell side and tubeside geometries of steam generators. These models will be employed in simulated sodium flow distribution tests.

In addition, since comparatively low pressure drops are desirable in heat transfer components to maintain the system pumping costs within economic bounds, efforts will be made to investigate shell side pressure drop.

Subtask 5.6.A.b - Heat Transfer and Water/Steam Distribution Test

The objective of this subtask is to determine heat transfer, pressure drop characteristics, and tube-flow distribution for various tube geometries, utilizing both scaled flow model tests and analytical methods.

In a once-through steam generator, water flowing through the tubes is converted into superheated steam. At some point along the tube, nucleate boiling process ceases (DNB), and an abrupt reduction in the heat transfer coefficient occurs. The position at which the change occurs is determined

by many factors including pressure, heat flux, mass velocity, tube shape and surface roughness. It is extremely important, therefore, to define the location of DNB so that the performance of the steam generator can be properly assessed and sufficient heat transfer surface provided. This test is required to determine the effect of the tube shape on the location of the DNB shape. Existing published information deals mainly with straight vertical tubes.

Two-phase pressure drop is also an area of uncertainty; a secondary activity of the experimental program would be to determine the pressure drop of the boiling fluid for the specific geometry.

Subtask 5.6.A.c - Thermal Baffle Development

This subtask will evaluate the behavior of conceptual thermal baffles in the steam generator under thermal transients, and the effectiveness of a design in reducing thermal stresses. Model thermal baffles will be designed, fabricated, and tested in a sodium loop to determine the thermal performance. The information obtained also would be used to develop a design basis for thermal baffle design, and therefore, be applicable to other components in the LMFBR, such as the intermediate heat exchanger.

Available data do not provide an adequate basis for specifying the baffles in the Westinghouse reference design.

Task 5.6.B - Development of Fabrication and Quality Control Technology

The objective of this task is to develop a fabrication, maintenance, and quality control technology consistent with the safety and reliability demands established by design criteria. Detailed manufacturing processes, fabrication techniques, and existing industry standards will be reviewed, developed, and revised, as required, to conform to the design, safety, and reliability criteria of the steam generator concept.

Development will concentrate on specific critical areas such as tube-to-tubesheet attachments, fabrication of tubesheets and tubes, thermal baffles, and quality control procedures and techniques (destructive and non-destructive). Efforts will be made to develop broad-based fabrication techniques and processes and, thus, to ensure a wide range

of applicability, independent of specific steam generators' designs.

Subtask 5.6.B.a - Welding Process Development

The objective of this subtask is to develop welding processes for reference materials and to provide welding specifications. Specifications will be evolved that will enable high-quality, reproducible welds for component fabrication compatible with the materials and designs specified by the other design and technical support programs.

Subtask 5.6.B.b - Tube-to-Tubesheet Joint Development

The objective of this subtask is the development of an optimum tube-to-tubesheet joint design having a high degree of reliability. Development activities will include:

- a. Evaluation of alternate joint designs for effective strength, weldability, and inspectability
- b. Development of fabrication methods, such as correct welding techniques and procedures, tolerances, and non-destructive test requirements
- c. Evaluation of the test welds in typical steam generator operating conditions, up to 2770 psi and 900°F
- d. Further evaluation of the tube-to-tubesheet joints under thermal shock and vibration to verify the analytical predictions and to provide a basis for selection of the jointure method.

The latter activity (item d -- transient testing of joints) will ensure a joint design that will have sufficient structural integrity to withstand the thermal gradients and stresses experienced by such incidents as sudden loss-of-feedwater or loss-of-sodium flow, and tube vibration. Tube-to-tubesheet joint mockups will be tested under subtask 5.6.C.b in a facility such as the Westinghouse General Purpose Sodium facility described in the Appendix.

Initial efforts will center upon attachment methods between Incoloy 800 tubes (~0.5 inch diameter x 0.065 inch wall) and Type 304 stainless steel tubesheet. These are the selected materials in the present steam generator concept. The design does not preclude, however, the use of more

suitable/economical materials if they become available.

Subtask 5.6.B.c - Fabrication/Manufacturing Process Development

This subtask will develop the fabricating techniques for the reference design and establish the most economical methods of manufacture allied to the required quality assurance standards.

To ensure the best possible heat exchanger requires 1) construction of representative model sections of the unit to develop the optimum fabrication procedures, and 2) assurance that the final article lends itself to acceptable inspection procedures to determine product quality. Detailed manufacturing clearances will be specified, assembly jigs and fixtures will be designed, inspection and testing procedures will be defined, and welder training and orientation programs will be prepared. The development of procedures which combine high quality design, fabricability, and effective inspection techniques will minimize the incidence of component failure.

In addition, specifications will be assembled from individual development tasks (i.e., tube-to-tubesheet, etc.) and will be assembled into one manufacturing process plan. This plan not only will describe the process but will also have quality assurance procedures and information regarding scheduling and cost estimating.

Subtask 5.6.B.d - Quality Control Development

The objective of this subtask is to develop quality control and inspection techniques and procedures consistent with the safety and reliability requirements established by the design criteria. The state-of-the-art will be reviewed and industrial practices will be assessed and revised as necessary. Since the goal of fabrication quality control is to detect, as early as possible, any substandard conditions that may arise, a coordinated activity, incorporating provisions for the periodic feedback of test data to the overall reliability assurance group, will be initiated to ensure proper dissemination and evaluation of latest information.

Subtask 5.6.B.e - Maintenance Development

The objective of this subtask is to a) define probable maintenance requirements, and b) develop and prove techniques, equipment, and procedures for inspection, repair, and retesting.

Even though the Steam Generator is the end-product of a comprehensive design program where operational reliability has been emphasized, provision must be made for maintenance. The heavy penalty and inconvenience of plant "down time" after attaining an operational status, makes necessary the development of proven maintenance procedures.

The problem involves:

- a. Adequate inspection techniques to prevent total failures.
- b. Defining the degree of deterioration before repairs must be made.
- c. Disassembly of the unit to the extent where the repair can be made.
- d. Testing the repaired area to ensure "as good as new" performance.
- e. Provisions for the repairing cleanup, reassembly, and verification of serviceability of the unit in the event of a Na/H₂O reaction.

If the current development of equipment, techniques and procedures can expedite any of the above areas and reduce the chance of a reoccurrence of this incident, then a worthwhile economic gain in plant utilization may be achieved.

Task 5.6.C - Mechanical and Structural Development

Subtask 5.6.C.a - Sodium/Water Reactions Test

The objective of this subtask is the development of techniques to predict the consequences of a sodium/water reaction in an operational steam generator for the full range of operating conditions. Experimental data will be obtained to enable analysis of the modes (i.e., rates of reaction, severity, etc.) of sodium/water reactions for the various tube configurations and tube materials. The correlation of experimental data with analytical techniques will provide the bases for development of analytical methods for evaluating effects of sodium/water reactions on tubes, internal structures, the shell, and the related heat transport system.

Test simulating defective welds and tube failures, at steam generator operating conditions, will be conducted in Westinghouse facilities.

Subtask 5.6.C.b - Development Testing

The objective of this subtask will be to proof test, under simulated environmental conditions and conditions of steady-state and transient operation, conceptual designs resulting from previous activities.

Included in this subtask will be proof testing of scaled models for:

- a. Tube-to-tubesheet joint design
- b. Thermal baffle designs
- c. Tube geometries and support designs.

Task 5.6.D - Follow-On Program Definition and Prototype Planning

The objectives of this task are 1) to identify and scope the additional research and development necessary to achieve safe, reliable, and economic sodium-heated steam generators, and 2) to plan for the design, fabrication, and test of steam generator models.

Subtask 5.6.D.a - Program Definition

The objective of this subtask is to identify and scope additional research and development activities. The basis for recommended research and development programs will be a review and evaluation of the results of earlier tasks, and of other AEC sponsored component development programs. A number of the programs are expected to be a continuation or expansion of the technical support programs, tasks 5.6.A, 5.6.B and 5.6.C. This subtask is in progress under AEC contract AT(30-1)4005.

Subtask 5.6.D.b - Prototype Planning

The objective of this subtask is to plan for the design, fabrication, and testing of steam generator models. Designs of "test models" will be prepared and will be of a size adequate for "proof-of-design" of the large components. The program to test these models will be scoped, assuming AEC facilities will be available for testing. The test program will cover both steady-state operation and transients testing for the system-imposed design limits.

Anticipated "Proof-of-Design" Model Steam Generators

<u>Thermal Capacity</u>	<u>Test Facility</u>
Number 1 - One Megawatt } Number 2 - One Megawatt } Number 3 - One Megawatt }	Facility such as <u>W</u> General Purpose Sodium Facility de- scribed in Appendix A
30 Megawatt	AEC-Sodium Components Test Installation
340 Megawatt	Demonstration Plant

Task 5.6.E - Model and Prototype Testing

The objectives of this task are to design, fabricate, test, and evaluate model steam generators. Conceptual design emerging from previous task will be detail-designed, according to design criteria and the functional requirements of the reference concept. Fabrication of models will utilize the latest developed specifications and techniques for welding and quality control.

Subtask 5.6.E.a - Model Tests (<1 MWt)

Models with capacities of less than 1 MWt will be designed and tested as preliminary proof-of-principle tests for various designs. A model between 1-2 MWt capacity will be designed and tested to incorporate many of the features of the 1300 MWt and 30 MWt designs. This model will serve to evaluate the chosen design on a small and relatively inexpensive scale before the design and fabrication of the 30 MWt unit is complete. Information gained in these tests will be factored into the design and used to specify the test program in the SCTI. Design and materials data derived from the test will also be used in a revised 1300 MWt design.

Subtask 5.6.E.b - Prototype Testing (30 MWt)

A 30 MWt prototype steam generator will be designed in such a manner

that its manufacture and planned test program will evaluate and demonstrate the principles of design, performance, and reliability required in a 1300 MWt unit. Various features of the 30 MWt prototype will be full-size duplicates, while other features will be scaled.

The 30 MWt prototype steam generator also must incorporate special test features so that the data needed for performance evaluation may be gathered. Accordingly, the unit will be designed with integral temperature, pressure, flow, liquid level, and strain-gage instrumentation.

Performance data applicable to the final design of a full-scale unit will include the following:

- a. Overall heat transfer performance at part and full load
- b. Fluid flow characteristics at part and full load
- c. Transient response behavior including flow stability
- d. Thermal and mechanical behavior of component parts under cyclic load

Subtask 5.6.E.c - 340 MWt SG -- Demonstration Plant.

An intermediate milestone toward the development of steam generators for a 1000 MWe plant will be the successful development of steam generators employed in a 300 MWe Demonstration Plant.

Task 5.6.F - Reliability Analysis

A principal objective of this task and of the overall program is to ensure that the developed heat transport component is reliable and safe. The design, fabrication, and testing of reduced-scale models must give assurance of long life for a nuclear power plant under a realistic pattern of operation. While economics is the common basis for the evaluation of all designs and experimental tests, reliability and safety will take precedence over economic optimization.

Activities to meet the task objective will involve the following:

- a. Coordination of Reliability Efforts
- b. Incorporation of Reliability in Design Reviews
- c. Specific Reliability Reviews
- d. Design of Test and Analysis of Results
- e. Fabrication Reliability

5.6.4 Relationship to LMFBR Program Plan

The AEC approach to sodium-heated steam generator development as described in Task Area 3-6, "Steam Generators," in Volume 3 of the LMFBR Program is in general agreement with the plan presented herein. The technical support activities coincide in overall program scope. A discrepancy may exist, however, 1) in the timetable for the testing of steam generator models and prototypes, and 2) in the availability of test facilities in the capacity range of 30 MWt or larger.

The Westinghouse schedule (Figure 5.6-2) calls for the testing of a 30 MWt steam generator to commence in the first quarter of 1972. If any delay is encountered in activities to provide new AEC test facilities, and with the demonstrated poor availability of the SCTI, it is quite possible that a conflict may arise in scheduling of the 30 MWt model of the W reference (1300 MWt) steam generator.

5.6.5 Major Test Facility Requirements

Major test facilities required for the development of steam generators are listed below.

Steam Generator Transient Facility (SGT)

The purpose of this facility is to test steam generator models under thermal transient conditions similar to those experienced in sodium-cooled fast reactor power plants. The SGT facility, consisting of a sodium system and a water system coupled by a steam generator model, will simulate the following transients: a) loss-of-feedwater pumping power and b) loss-of-secondary pumping power.

These are the most severe transients to be expected, producing temperature variations of about 30°F/second to 40°F/second.

This facility will also be capable of imposing transients sufficient enough to simulate effects of extended life on the steam generator model.

1 MW Sodium/Water Test Facility

This purpose of this facility is to provide for the testing of sodium-heated steam generators with thermal capacities to 1 MWt. The sodium system should be capable of providing:

Flow	200 gpm
Temperatures to	1200°F
Heat Input	1 MWt

The purpose of the water/steam system is to provide feedwater to steam generator models at a range of controlled pressures and temperatures and to condense the superheated steam received from the models. The loop will also be required to simulate some of the transients that can be experienced in full-size steam generators. It should contain sufficient equipment to duplicate a range of operating water chemistry requirements. Conditions for the water/steam loop should provide:

Feedwater flow	400 - 3750 lb/hr
Maximum feedwater temp.	650°F
Feedwater pressure	2000 - 4000 psi
Heat dump capacity	1 MW

Some of the steam generator auxiliary systems to be incorporated in this facility are:

- a. The sodium shut-off and dump system with dump tank
- b. The water shutoff and dump system with dump tank
- c. The blow-off system with blow-off tank
- d. The hydrogen detection system with hydrogen detector vapor traps and freeze traps
- e. An argon cover gas system with vapor trap

The Westinghouse equivalent of this facility is described in the Appendix as the combination of the Steam/Water Loop (SWL-1) and the General Purpose Sodium Loop-1 (GPL-1).

Sodium-Water Reaction Test Facility (SWR)

The purpose of this facility is to perform tests and provide experimental data resulting from the reaction of sodium and water under conditions experienced in sodium-heated steam generators.

The SWR Facility must be capable of monitoring and recording variables with rapid response and the desired accuracy of measurement as follows:

- a. Pressure: 0-1000 psig in liquid sodium at reaction point
- b. Temperature: 0-1500°F in liquid sodium at reaction point
- c. Flow Rate: Water - Up to 16 gpm @ 2700 psig
Sodium - Up to 2100 lb/hr @ 1200°F
- d. Hydrogen Concentration
Cover Gas - 10 ppm to 100 percent
Dump Tank - 10 ppm to 100 percent
- e. Necessary variables to control loops and system integrity.

The Westinghouse Sodium Water Reaction Facility (SWR-1) is described in Appendix A.

5.6.6 Program Schedule

The steam generator development schedule is shown in Figure 5.6-2. A task-by-task description of the illustrated milestones is presented below.

Task 5.6.A - Thermal and Hydraulic Development

- 1. Preliminary Test Data Evaluation (1 MWt Model)
- 2. Data Analysis (1 MWt Model) Complete
- 3. Thermal Baffle Development Complete
- 4. Analysis of Preliminary 30 MWt Data

Task 5.6.B - Fabrication and Q.C. Technology

- 1. Input -- State-of-the-Art/Current Technology
- 2. Preliminary Fabrication and Quality Control Specs Complete
- 3. Development of Tube/Tubesheet Attachment Complete

4. Final Fabrication and Q.C. Specs Complete

Task 5.6.C - Mechanical and Structural Development

1. Start -- Testing of Sub-Components
2. 1st Phase Sodium/Water Reaction Tests Complete
3. Data Analysis Complete -- (1st Phase Sodium/Water Reaction Tests)
4. Sub-Component Assembly Tests Complete

Task 5.6.D - Follow-on Program Definition and Planning

1. Identification of Follow-on Tasks

Task 5.6.E - Model and Prototype Testing

1 MWt Model

1. Start Model Testing
2. Complete Model Testing

30 MWt Prototype

3. Start -- Model Design
4. Order Materials
5. Complete Fabrication
6. Start Model Testing
7. Complete Model Testing

~340 MWt - Demonstration Plant Steam Generator

8. Start 340 MWt Demo Plant Steam Generator Design
9. Order Materials
10. Place Order Demo Plant Steam Generator
11. Analysis of Preliminary 30 MWt Data
12. 340 MWt Steam Generator Startup

1300 MWt - 100 MWt Plant

13. Start 1300 MWt Steam Generator Design

Task 5.6.F - Reliability Analysis

1. Preliminary Procedures and Methods for Reliability Analysis

Figure 5.6-2. Steam Generator Development Program Schedule

5.7 Fuel Handling and Maintenance Systems Development

5.7.1 Objective

To develop components and subsystems for a hot cell refueling system to a level consistent with safety, economic, and reliability goals for the 1000 MWe LMFBFR plant.

5.7.2 Program Summary

The program is divided into fourteen tasks. The first eight tasks are devoted to development of electrical and mechanical components and materials. The last six tasks are devoted to development of major subsystems culminating in full scale proof testing of prototype fuel handling system equipment. The last six tasks are dependent on information generated during the first eight. These tasks are listed below and described in the next section.

- Sodium Vapor Deposition Studies
- Bearing Development
- Mechanisms Development
- Seal Development
- Electrical Component Development
- Materials Development
- Lubricant Development
- Under Sodium Scanning Development
- Development of Cooling for Fuel Handling Systems
- Manned Access System Development
- Viewing System Development
- Manipulator System Development
- Prototype Fuel Handling Machine Development
- Shipping Cask Development

5.7.3 Task Descriptions

Task 5.7.A - Sodium Vapor Deposition Studies

The objectives of this task are to develop methods for predicting evaporation rates of sodium from open pools and to determine the

necessity and feasibility of using devices to limit this evaporation. Excessive evaporation of sodium and subsequent deposition from the reactor and spent fuel storage pits is potentially a serious problem in hot cell refueling. Potential consequences of this problem include: sticking and galling of mechanical components; short circuiting of electrical components; deterioration of optical surfaces; and cell contamination by sodium 22 and 24. In this task, open pools of sodium will be exposed to inert atmospheres. Sodium buildup will be monitored for various pool diameters and sodium temperatures. The effects of cover gas flow velocities and patterns will be considered. Analysis will be performed to develop empirical methods for predicting sodium evaporation and deposition rates for generalized cases.

Task 5.7.B - Bearing Development

The objective of this task is the development of bearings for LMFBR fuel handling systems. Of particular concern are the bearings associated with the lateral fuel handling machine which must support loads of up to 1000 lb and operate reliably under sodium. In this task, bearing test rigs will be designed which will permit operation of prototype sleeve and rolling element bearings under sodium and in inert gas/sodium vapor environments. Sodium and inert gas purities will be controlled to determine the effects of oxygen levels on bearing behavior. Friction and wear rates will be measured and the effects of bearing stresses on specific wear rates will be determined. The effects of intermittent operation will be assessed. Initial tests will be used to establish bearing designs for refueling equipment. Subsequent tests will be devoted to the determination of bearing life for extended operation.

Task 5.7.C - Mechanisms Development

The objective of this task is to develop mechanisms for LMFBR hot cell refueling equipment which will operate reliably in sodium and inert gas-sodium environments. Mechanisms to consider include vertical fuel handling machine grippers, lateral fuel handling machine basket releases,

roller chains, gears, racks, and linkages. Screening and life tests will be performed under prototypical conditions in specially designed test chambers.

Task 5.7.D - Seal Development

The objective of this task is to develop seals for LMFBR hot cell refueling systems. Inert gas filled hot cells require a high degree of seal reliability. Depending on cell pressures, leakage of contaminated gas could create a severe hazard. Leakage of oxygen into the hot cell, on the other hand could lead to contamination of sodium surfaces and/or excessive cell atmosphere purification costs. Seals for various applications will be tested under prototypical conditions. These seals include those for: viewing windows, electrical and gas penetrations; personnel and equipment access hatches; spent fuel and contaminated equipment storage pits; and personnel access suits and systems.

Task 5.7.E - Electrical Component Development

The objective of this task is to develop electrical components for use in LMFBR hot cell refueling systems. Acknowledged problems involved with operating electrical equipment in inert atmospheres include: electrical breakdown caused by lower dielectric strengths; overheating caused by poor heat transfer properties; accelerated brush wear on commutating equipment; and sticking and pitting of contacts. Additional difficulties may be introduced by the presence of sodium vapor. In this task electrical components previously screened for radiation resistance, such as motors, position sensors, relays, etc., will be tested in an inert gas/sodium vapor environment. In addition, more fundamental tests will be performed to determine voltage and spacing limitations on electrical conductors in a hot cell environment.

Task 5.7.F - Materials Development

The objective of this task is the selection and development, if required, of miscellaneous materials for LMFBR hot cell refueling applications. Such materials include: glass for viewing systems; paints and coatings for refueling cell walls and equipment; fabrics

for personnel access suits; and various plastics and elastomers used on equipment. Candidate materials will be carefully screened to determine if there are any known reactions or instabilities which could occur in service. Samples of the most promising materials will be exposed to an argon/sodium vapor environment for extended periods of time. They will be removed and tested periodically to determine if deterioration has occurred and to what degree.

Task 5.7.G - Lubricant Development

The objective of this task is to select and develop, if required, lubricants for LMFBR hot cell refueling system applications. Candidate lubricants will be selected on the basis of high radiation resistance and low vapor pressure. Procedures will be established and tests performed to evaluate lubricant load capacity and performance in argon/sodium vapor atmospheres.

Task 5.7.H - Under Sodium Scanning Development

The objective of this task is to develop a reliable method of scanning the area beneath the surface of a molten pool of sodium, which is opaque to devices that function in the visible spectrum. The most promising possibility is a modified version of the common "SONAR" devices that utilize a supersonic sound wave for the probing energy. The primary design task will be to design a transducer that can survive in a sodium environment. After a suitable transducer is developed, a system will be engineered to best utilize its potential. The systems that can be investigated are those that use multiple arrays of transducers, special geometry transducers, or elaborate scan-pan system utilizing one transducer. A component level test program will be performed to verify which system should be expanded. After a "best" system solution is reached, a performance test program must be run under actual environment conditions to verify the reliability of the entire system.

Task 5.7.I - Development of Cooling for Fuel Handling Systems

The major objectives of this task are to establish design concepts that will eliminate the possibility of fuel melting during transfer under all

possible operational malfunctions, and to develop systems and components capable of long-term reliable cooling operation in the sodium vapor-inert gas environment. Detailed design concepts for cooling systems will be prepared. These concepts will examine the advantages and disadvantages of different coolants (argon, helium, sodium, NaK, etc.) and operating parameters. Available components for gas, and liquid metal circulation systems including devices for de-entrainment of sodium droplets will be examined. Components will be provided and tests will be performed on a prototype loop assembly. During the tests, abnormal conditions, such as "sodium frosting", and malfunctions will be observed. Data will be evaluated and deficiencies of the design delineated.

The results of the above evaluation will serve as guideline requirements for either modifying existing equipment or design of new component applications. The design will be modified as necessary to reflect the effects of the modified and new components and will select those components offering the most potential for successful performance. The selected devices will be fabricated and again screened. On the basis of test results, the most promising coolant system concepts will be prototyped and subjected to extended life tests to accumulate adequate statistical data for meaningful reliability analysis.

Task 5.7.J - Manned Access System Development

The objective of this task is to provide a proven Manned Access System to permit contact maintenance in inert cells. Development of the Manned Access System will be broken down into four major areas:

- a. Environmental suits including body coveralls, gloves, boots, and helmets
- b. Supply and return breathing and cooling air systems, including compressors, filters, driers, piping, vacuum pumps, flexible hoses, and air regulation equipment
- c. Access locks, including doors and associated controls, inert gas suit purging system, change room, suit cleaning facilities, storage racks

- d. Emergency equipment including emergency breathing apparatus, suits, and apparatus for emergency rescue entry, etc.

Once the location of equipment in the hot cell and maintenance areas is known and the maintenance tasks are defined, a basic suit will be designed. Before fabrication of the suit, materials compatibility tests of candidate suits, boot, glove, and helmet materials must be conducted in a sodium vapor/dry argon atmosphere. Selected material must then meet Federal Specifications GGG-H-142e and CCC-T-191b and Military Specification MIL-C-19002B. After construction and leak checks, the suit will be coupled to the supply and the return breathing and cooling air system, and air trials initiated. Upon successful completion of air trials, testing in a hot cell mockup will begin. Little development work is required for the remaining three sub-system areas; most components required are available off-the-shelf. Testing of each subsystem will be required, however, upon completion of fabrication.

Task 5.7.K - Viewing System Development

The objective of this task is to develop better equipment for viewing into high radiation areas, such as hot cells, when direct vision is not possible. Glass which is more resistant to radiation will be investigated in an attempt to obtain a type that will provide more attenuation to radiation with less light attenuation. An attempt will be made to improve the aging characteristics; that is, the non-browning when exposed to high radiation fields. Then the glass will be tested in applications for lenses for the television cameras and telescopes, as well as the shielding windows. The television camera, in addition to the lenses already mentioned, will be developed more exactly for high radiation, and hostile environment usage. The actual in-cell circuitry will be reduced to an absolute minimum. Operating voltages will be lowered as much as possible. Remote maintenance and repair will be advanced as much as possible. Allowable temperature rise within the camera will be raised as much as possible to possibly eliminate the need for additional cooling. Cooling techniques will be developed to allow operating in elevated inert atmospheres when self-cooling is inadequate. Test

programs will be established for all items to verify reliable operation in the expected environment.

Task 5.7.L - Manipulator System Development

The objective of this task is to provide suitable manipulator systems for remote maintenance in the hot cell and maintenance areas. When radiation levels are high or when manned access is not practical, manipulator systems will be required to do maintenance work. Although it is generally felt that no extensive manipulator design development programs will be necessary, information on how the dry argon/sodium vapor environment will affect the lubrication cooling and electrical properties of manipulator components is required. The extent of remote maintenance operations and the capabilities of all manipulators must be known before a selection can be made and a maintenance program thoroughly defined. After fabrication, the selected machine(s) will be tested under actual conditions in a cell mockup.

Task 5.7.M - Prototype Fuel Handling Machine Development

The objective of this task is to perform full-scale development testing of prototype fuel-handling machines. Prototype fuel-handling machinery will be designed, constructed, and operated in a prototypical environment. The design effort will utilize information developed during component level testing, Tasks 5.7.A - 5.7.G to the greatest degree possible.

Task 5.7.N - Shipping Cask Development

The objective of this task is to develop a shipping cask, for spent LMFBR elements, capable of minimizing the effect of the cost of shipping and cooling on the fuel cycle cost. Since thermal reactor casks are not suitable for shipping fast reactor fuel elements, a new technology in shipping cask design must be developed. Alternate cask shielding materials, such as steel and depleted uranium, must be evaluated. LMFBR fuel probably will require a shipping cask containing separately canned assemblies. These steel cans may be filled with a liquid metal to help dissipate the decay heat better and more evenly throughout. In the case of lead alloys, they possibly also may provide some shielding and

additional safety containment. The cans then would be placed in a cask cooled by some organic liquid. Development tasks include:

- a. Can design
- b. Cask design, including a study of the optimum cask shape and number of cans/cask
- c. Cask and can materials selection
- d. Cask and can fabrication techniques
- e. Primary and secondary coolant selection

After selection of a design, a scale model will be fabricated and tested to the hypothetical accident conditions specified in 10-CFR-71.

5.7.4 Relationship to LMFBR Program Plan

Many of the required programs are contained in the LMFBR Program Plan. Bearing, mechanism, seal, electrical component, gas cooling system, and manipulator development programs are covered in Task Area 3-7, "Core Components Handling Equipment," in Volume 3. This section also covers full-scale testing of prototype fuel handling machines. Volume 7, "Fuels and Materials" is written in sufficient generality to include materials and lubricant development within its scope. Reinforcement of the Program Plan, however, is required in the following areas:

- Task 5.7.A - Sodium Vapor Deposition Studies
- Task 5.7.H - Under Sodium Scanning Development
- Task 5.7.J - Manned Access System Development
- Task 5.7.K - Viewing System Development
- Task 5.7.N - Shipping Cask Development

5.7.5 Major Test Facility Requirements

Environmental Test Chambers (Tasks 5.7.A - 5.7.G)

Approximately four to six environmental test chambers are required to carry out the component level testing described in Tasks 5.7.A - 5.7.G.

Each of these environmental chambers is, in effect, a small reactor and hot cell replica consisting of an inert gas chamber containing an open pool of sodium. Heating/cooling, circulation and purification provisions will be required to maintain prototypical conditions as follows:

Inert gas purity	1-100 ppm, H ₂ O and O ₂
Inert gas temperature	70-150°F
Sodium purity	1-100 ppm, O ₂
Sodium temperature	250-600°F

In addition to the test chambers, loading devices and fixtures will be required to provide the necessary loads and motions for component testing. Westinghouse facilities of this type are described in the Appendix (Sodium Vapor Test Facility, HOCMUF, MUST, GBET)

Large Sodium Pool (Task 5.7.H)

The major facilities requirement for development of under sodium scanning devices is a large pool of sodium, perhaps 6-10 ft. in diameter and 15-20 ft. deep. For fuel handling systems development, the pool should be capable of being heated to 600°F. However, a temperature capability of 1200°F would be desirable since this would permit testing at reactor operating temperatures. Besides the sodium pool, inert cover gas systems, sodium purification systems, test instrumentation, etc. will be required. As an alternate to constructing a facility specifically for scanning devices, it may be possible to use the environmental chambers specified for Tasks 5.7.A - 5.7.G for preliminary transducer development and to perform full-scale system testing in the reactor hot cell mockup specified for Tasks 5.7.J - 5.7.M.

Heat Transfer Loops (Task 5.7.I)

Forced convection inert gas and liquid metal loops are required which can accept test sections up to 10 inches in diameter by 12 ft. long. They will provide heating/cooling rates of up to 50 KW and permit hot section

temperatures of up to 1600°F. The gas test loop must also have the capability of introducing specimens contaminated with sodium and de-entraining sodium particles from contaminated specimens during operation. The loop purification systems must maintain O₂ and H₂O levels in the ppm range. The Westinghouse Spent Fuel Assembly Heat Transfer Test Facility is described in the Appendix.

Refueling Hot Cell Mockup (Tasks 5.7.J - 5.7.M)

A large-scale mockup of a refueling hot cell and reactor vessel is required for prototype testing of manned access systems, manipulators, viewing systems and fuel handling machinery. For the Westinghouse 1000 MWe reactor a cell approximately 40 ft. long by 25 ft. wide by 50 ft. high is necessary, and containing an 8 ft. diameter, 30 ft. deep, simulated reactor vessel. These cell and vessel sizes will permit full-scale testing of the vertical fuel handling machine. A significantly larger cell and vessel is required if a full scale testing capability for the lateral fuel transfer machine is included. The mockup cell should have the capability of being inerted and the reactor vessel the capability of being filled with sodium. Heating/cooling, circulation, and purification systems for the sodium and argon will be required to keep temperature and purities at prototypical levels. Additional service and test equipment will be required including a bridge crane, communication systems, test instrumentation, a dummy reactor core, etc. (The Westinghouse Fuel Handling Test Facility, shown in the Appendix will meet these requirements.)

Materials Test Loops (Task 5.7.N)

A materials testing facility will be required to run coolant-structural materials, compatibility tests, and proof-of-principle tests on fabrication techniques. Facilities must be provided for testing the scale model to the hypothetical accident conditions for shipping casks set forth in 10-CFR-71. This facility will provide a non-yielding horizontal dropping pad, fire test area (1475°F fire for 30 minutes) and an immersion pool.

5.7.6 Program Schedule

The development schedule for fuel handling and reactor plant maintenance systems is shown in Figure 5.7-2. A task-by-task description of the illustrated milestones is presented below:

Task 5.7.A - Sodium Vapor Deposition Tests

1. Complete component level screening tests

Task 5.7.B - Bearing Development

1. Complete component level screening tests
2. Complete component level life tests

Task 5.7.C - Mechanism Development

1. Complete component level screening tests
2. Complete component level life tests

Task 5.7.D - Seal Development

1. Complete component level screening tests
2. Complete component level life tests

Task 5.7.E - Electrical Component Development

1. Complete component level screening tests
2. Complete component level life tests

Task 5.7.F - Materials Development

1. Complete component level screening tests

Task 5.7.G - Lubricant Development

1. Complete component level screening tests

Task 5.7.H - Under Sodium Scanning Development

1. Complete subsystem level tests
2. Complete proof testing of Demonstration Plant equipment

Task 5.7.I - Cooling System Development

1. Complete subsystem level tests
2. Complete proof testing of Demonstration Plant equipment

Task 5.7.J - Access System Development

1. Complete subsystem level tests
2. Complete proof testing of Demonstration Plant equipment

Task 5.7.K - Viewing System Development

1. Complete subsystem level tests
2. Complete proof testing of Demonstration Plant equipment

Task 5.7.L - Manipulator System Development

1. Complete subsystem level tests
2. Complete proof testing of Demonstration Plant equipment

Task 5.7.M - Prototype FHM Development

1. Begin construction of fuel handling test facility
2. Complete fuel handling test facility
3. Complete subsystem level tests
4. Complete proof testing of Demonstration Plant equipment

Task 5.7.N - Shipping Cask Development

1. Complete subsystem level tests
2. Complete proof testing of Demonstration Plant equipment

TASK	TITLE	CALENDAR YEAR											
		1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979
5.7.A	Na VAPOR DEPOSITION TESTS		1			2							
5.7.B	BEARING DEVELOPMENT			1		2							
5.7.C	MECHANISM DEV.			1		2							
5.7.D	SEAL DEV.			1		2							
5.7.E	ELECTRICAL COMP. DEV.			1		2							
5.7.F	MATERIALS DEV.			1									
5.7.G	LUBRICANT DEV.			1									
5.7.H	UNDER SODIUM SCAN. DEV.					1		2					
5.7.I	COOLING SYSTEM DEV.					1		2					
5.7.J	ACCESS SYSTEM DEV.					1		2					
5.7.K	VIEWING SYSTEM DEV.					1		2					
5.7.L	MANIPULATOR SYSTEM DEV.					1		2					
5.7.M	PROTOTYPE FHM DEV.		1		2	3		4					
5.7.N	SHIPPING CASK DEV.					1		2					

Figure 5.7-2. LMFBR Fuel Handling Development Schedule

6. INSTRUMENTATION AND CONTROL DEVELOPMENT

6.1 Instrumentation Development

6.1.1 Objective

To develop instrument technology to a level where sensor devices can be manufactured to meet the functional requirements of the reference 1000 MWe LMFBR plant without undue technical and economic risk.

6.1.2 Program Summary

The program is divided into tasks delineated by the type of plant variable being monitored. Each task program is complete within itself, including the development and testing of alternative devices. There is some overlap between the tasks in the region of materials development and the testing facilities required. Since all instruments require some means of transmitting their signals, all signal transmission components have been included in a separate task. The nine tasks comprising this program are listed below and are described in the next section.

- Nuclear Instrumentation Development
- Temperature Instrumentation Development
- Coolant Flow Instrumentation Development
- Pressure Instrumentation Development
- Strain Measurement Gauge Development
- Level Measurement Instrumentation Development
- Fission Product Detection Instrumentation Development
- Leakage Detection Instrumentation Development
- Signal Transmission Components Development

For all tasks, it is assumed that proof tests of instruments and instrument systems will be performed in the FFTF and the Demonstration Plants.

6.1.3 Task Descriptions

Task 6.1.A - Nuclear Instrumentation Development

The objective is to develop reliable and economic radiation measuring instruments capable of providing signals to the control and safety circuits under all reactor operating conditions from in-vessel and neutron-shield environments. This task is divided into five subtasks covering the different instruments required and electronic circuits. The cables are included in Task 6.1.I.

Desirable characteristics for nuclear instrumentation in general are:

a) a minimum life of at least one year, b) rapid response, and c) high accuracy. Failure to achieve these characteristics and those which may be particular to a given instrument will result in an economic penalty due to increasing the margins to failure for safety reasons.

The function of the nuclear instrumentation is to generate signals proportional to the neutron or gamma flux at the instrument site over those power ranges for which the instrument was intended to operate and to transmit the signals to an electronic counter which indicates the site flux.

Subtask 6.1.A.a - Low Power Neutron Fluxmeter

The objective of this subtask is to develop fission chambers capable of operation in the lower axial blanket of an LMFBR. In addition to the required characteristics as listed in the Information Needs Tables, Section II of this report, this instrument must be mechanically robust in order to withstand movement in a thimble of limited curvature on insertion and retraction. Failure to achieve the above will result in increased instrument replacement costs. Failure will also cause loss of accuracy and loss of ability to detect fault conditions early during startup and shutdown, unless methods can be developed using sites more distant from the core although more accessible for maintenance.

No instruments are currently available for operation at these conditions although some are under development in USA, UK, and elsewhere. Specifically, the studies required include: a) specification of the neutron

should suffice to complete the development program. However, if locations cannot be found at EBR-II with a sufficiently high gamma/neutron ratio, then, tests at FFTF may be required.

Subtask 6.1.A.c - Primary Sodium Gamma-Activity Monitor

The objective of this subtask is to develop ion chambers suitable for gamma monitoring while located in the cover gas blanket. The required characteristics of the ion chambers are as given in the Information Needs Tables in Section II of this report. Failure to develop suitable instruments would lead to alternative concepts; namely, the use of closed thimbles or an out-of-vessel monitoring station. Current LMFBR practice is to protect all in-vessel instruments in cooled thimbles.

The studies include: a) economic assessment of the advantages and disadvantages of the three concepts above, including the costs of the development required, b) specification of the normal and fault gamma flux levels in the cover gas, c) extension of Subtask 6.1.A.a. to gas blanket ion chamber requirements, and d) lifetime and sensitivity testing of prototypes under operating conditions.

Since the sodium activity monitor is an in-vessel instrument, its development will follow closely the program for the low power neutron flux-meter, namely, bench tests, non-active sodium environment tests, and reactor tests in FFTF.

Subtask 6.1.A.d - Electronic Equipment

The objective of this subtask is to develop electronic equipment capable of counting pulses from nuclear flux detectors at a rate of at least 10^6 counts/second under practical circumstances. Failure to achieve this count rate will result in a requirement for additional flux measurement channels or in reduced monitoring sensitivity. Counters now available are capable of 10^5 counts/second under operating conditions and 10^7 cps under bench conditions with short cable lengths.

Specifically, the studies required include: a) instrument design and logic development for wide-range and high-rate counting against monitor

high gamma backgrounds and b) testing of prototypes in the correct environment for sensitivity, accuracy, and reliability.

The development of the electronic circuit associated with the nuclear instruments can be undertaken by the instrument vendors with on-site testing in association with the reactor tests of the instruments. Tests of wide-range channels are currently being carried out at ANL.

Task 6.1.B - Temperature Instrumentation Development

The objective is to develop temperature sensing devices capable of operation in high temperature liquid metal environments. The required characteristics are given in the Information Needs Tables. The current state of technology is such that failure to achieve a suitable device is not likely except that instrument life may be unsatisfactory. Failure to achieve the desired life will result in economic penalties due to increased replacement costs and the need to incorporate facilities for easy replacement either on- or off-load.

Subtask 6.1.B.a - Thermocouples

The objective for this subtask is to develop thermocouples suitable for in-vessel applications. Due to the advanced status of technology in this area, the main study consists of the testing of statistically significant numbers of samples under LMFBR conditions for accuracy, sensitivity, and reliability under steady-state and thermal cycling conditions. Other specific studies include maintenance and replacement techniques for exact positioning of the thermocouple, testing on a mockup subassembly to determine the effect of position upon the temperature indicated, and the development of a suitable cold junction box.

The testing of thermocouple samples currently is on-going at the National Laboratories. The development of maintenance and location techniques can be undertaken by the reactor designer.

and gamma fluxes at the instrument site, b) determination of the optimum materials and designs for the instrument (including chamber body, neutron sensing medium, gas fill) and gamma tolerance and the effect of neutron spectrum, c) cabling (see Subtask 6.1.I.b) and d) lifetime and sensitivity testing of prototypes under operating conditions.

The development program required may be based upon instrument vendor experience, e.g. Westinghouse Electronic Tube Division (WETD), and the prototype instruments tested successively in bench rigs, sodium loops as at ANL, and reactor facilities, such as FFTF, to obtain sufficient gamma-background.

Subtask 6.1.A.b - Intermediate and High Power Neutron Fluxmeters

The objective of this subtask is to develop fission and ion chambers for use in the radial neutron shield. Since the instruments for the intermediate and high-power ranges are similar, they have been combined into a single subtask. Their functional characteristics are as specified in the Information Needs Tables. Failure to develop the required instruments is not expected since EBR-II and Fermi use this out-of-vessel instrumentation concept and the instruments themselves are only minor extrapolations of existing instruments. The modular core concept, however, may prove to be unamendable to out-of-vessel instrumentation, in which case, the development studies of Subtask 6.1.A.a would be extended to include the environmental conditions at alternative in-vessel sites for intermediate and high-power instruments.

The studies required include: a) feasibility of using out-of-vessel instrumentation to monitor a modular core, b) specification of fluxes at instrument site, c) lifetime and sensitivity testing of instruments including cooled and uncooled concepts and determination of the optimum form of gamma compensation, and d) the extension of the low power instrument program to cover in-vessel ion chambers if the need is indicated by study, a) above.

For these out-of-vessel instruments the experience of vendors from current thermal nuclear power stations and tests at the EBR-II facility

Subtask 6.1.B.b - Alternate Temperature Sensing Devices

A thermocouple is not considered a temperature standard, hence, the objective of this subtask is to develop alternate devices which may be used either as standards or as back-up instruments.

The resistance thermometer is a standard whose range is being extended to LMFBR conditions. Many of the problems associated with it are identical with those of thermocouples and are being solved simultaneously, e.g., materials and compatibility, gamma heating. Other methods include ultra-sonic devices and thermistors, although considerable development is required for these. Failure to develop these instruments will have little effect on the program unless the thermocouple program fails.

The specific studies required include: a) performance and lifetime testing of prototype resistance thermometers, b) development of materials and designs for thermistors, and c) feasibility and economic studies of ultra-sonic methods to justify the development costs.

The development of resistance thermometers can be aided by borrowing from the thermocouple program. Industrial laboratories with experience in high temperature technology (e.g. Westinghouse Astro-Nuclear Laboratory) can provide facilities for the initial design. A statistical, sample-testing program by the National Laboratories for comparison with the thermocouple program will follow.

Task 6.1.C - Coolant Flow Instrumentation Development

The objective of this task is to develop instruments capable of measuring the flow in the pipework of an LMFBR. The required characteristics are given in the Information Needs Tables. There are several forms of devices available, the chosen one requiring least basic development. In view of the alternatives available, failure to achieve a suitable device is not expected.

Subtask 6.1.C.a - Venturi

The object of this subtask is to develop a Venturi and associated instrumentation suitable for measuring liquid sodium flow rates of 120,000 gpm

in a pipe 4 feet in diameter. The basic device, the Venturi, is well understood, and the associated circuitry for measuring the throat pressure drop using e-m pump and flowmeter is within the present state-of-the-art. Specific studies include: a) tests on a large scale device and b) the effect on calibration of variations in pipe circuit configuration.

Due to the size of the Venturi, all tests and calibration will require the use of LMEC facilities, e.g. SPTF.

Subtask 6.1.C.b - Alternative Flow Measuring Instruments

The Venturi is an expensive instrument due to its size and facility requirements for its calibration. Furthermore, its accuracy is rather poor, about $\pm 5\%$. Alternative methods use turbines, thermal transit times, eddy currents, and e-m systems; the preferred alternative uses the leading edge of an ultra-sonic signal. Such an instrument is relatively small and cheap. The required studies in this subtask include: a) the acoustic path design studies including transducer cooling and attachment, electronic circuitry and calibration methods, b) magnetic materials and coil development, and c) bearings and calibration problems.

The development of suitable alternative flowmeters can be undertaken by industrial concerns with expertise in the respective areas (e.g. Westinghouse Research and Development has considerable experience with ultra-sonic techniques). Small-scale prototypes can be tested in loops (e.g. GPL-2) and applied at LMEC facilities.

Task 6.1.D - Pressure Instrumentation Development

The objective of this task is to develop instruments to measure absolute and differential pressures in an LMFBR. In view of the success of current devices, failure is unlikely. The required capabilities for these instruments is given in the Information Needs Tables in Section II of this report.

Subtask 6.1.D.a - Absolute Pressure Measurement

Most measurements of absolute pressure employ a diaphragm or bellows separating the primary fluid from a pressure-transmitting secondary fluid. Although gas has been used, the secondary fluid usually is NaK. The objective of this subtask is to develop a satisfactory pressure transducer and transmission line (see Subtask 6.1.I.C). The required studies include: a) unifying existing data and analysis of the failures of instruments in existing plants and b) the testing of the prototype developed using this data.

The development of improved instruments from existing models can be carried out by industry, and the testing of large samples performed by the National Laboratories.

Subtask 6.1.D.b - Differential Pressure Measurement

The objective of this subtask is to develop devices to measure differential pressure in an LMFBFR. There are two methods available within the scope of existing technology. One employs an e-m pump to create a null flow condition in a bypass line and an e-m flowmeter to detect the condition. The power required by the pump can be related to the pressure difference. The other method uses a suitable, adapted Barton differential pressure cell. The work required by this subtask includes: a) adaption of devices from Subtask 6.1.C.a (flow measurement) to differential pressure indication and b) test data from these devices and Barton cells.

Since little or no basic research is required, the development program for differential pressure monitors will follow the same path as for absolute pressure transducers.

Task 6.1.E - Strain Measurement Gauge Development

The purpose of this task is to develop strain gauges for operation on an LMFBFR plant. Strain gauges are unlikely to be developed for use in contact with sodium before a 1000 MWe LMFBFR is committed. Gauges capable of use on the outside of the plant and components have been reported although gauge life is only moderate. The critical, functional

requirements are given in the Information Needs Tables. Failure to achieve suitable gauges will result in over-conservative plant designs.

The studies required include: a) continued testing and development of promising gauge designs, b) investigations into new methods of attachment, and c) a feasibility study into new methods of strain measurement.

The development program for strain gauges can be based on the existing experience of industrial vendors in association with such programs as NERVA at Westinghouse Astro-Nuclear Laboratories and LMEC test programs.

Task 6.1.F - Level Measurement Instrumentation Development

The purpose of this task is to develop devices suitable for measuring liquid sodium levels in an LMFBR. The functional requirements and state-of-the-art are given in the Information Needs Tables. Experience with existing units indicates that failure to achieve satisfactory devices is unlikely. However, due to uncertainties regarding phenomena related to wetting and impurities, it is suggested that at least two instruments operating on different principles be employed.

Subtask 6.1.F.a - Inductance Probe

The objective is to develop an inductance probe and associated circuitry capable of measuring sodium levels. Required studies include: a) development of materials and coils for use up to 1200°F, b) study of wetting and impurity phenomena in relation to probe performance, and c) testing of a statistically significant sample of devices.

The development of improved probes can be undertaken by existing vendors, and the test program for evaluating the different models can be performed by the National Laboratories using FFTF for the environmental testing of in-vessel instruments.

Subtask 6.1.F.b - Alternative Devices

The objective is to develop alternative level measuring devices operating on principles different from the inductance probe. Several basic methods have been developed including floats, bubblers, resistance probes,

differential pressure devices, ultra-sonics, sodium activity, and micro-waves. Studies required include: a) assessments of alternative schemes and the selection of those requiring least development to attain suitable operational status, and b) performance and lifetime testing of a requisite number of samples of each device.

The development of instruments operating on different principles should be performed by laboratories with expertise in the respective areas (e.g. Westinghouse Research and Development has experience in ultra-sonics). Comparative testing of different probes would be done by the National Laboratories.

Task 6.1.G - Fission Product Detection Instrumentation Development

The goal of this task is to develop methods for detecting fission products indicative of failed fuel rods. The problems are associated with Na 24 and the activity of fission products released in a vented fuel design. Methods of analysis to indicate fission products are available, but the delay time is long and methods of detecting the faulty subassembly inadequate. Failure to achieve a rapid response system and quick fault location is possible. The penalties associated with failure are: a) above normal radioactivity levels in the primary circuit and b) possible gross fuel damage, voiding, and local melting.

In order to detect changes in fission product concentration in the sodium or cover gas from those levels associated with the normal leakage from vented fuel, the following studies are required: a) a study of fission product levels and distribution in the primary sodium and cover gas for closed and vented fuels and their relation to irradiation and refueling, b) a study of existing methods of fission product detection and refinements of the most promising, c) a comparison study of efficiencies and responses of liquid and gas analysis systems, d) a feasibility study for different sampling and location schemes, e) a design study for selected sampling methods, f) development and testing of detection system circuitry, g) sensitivity testing of detection system on an operating reactor, and h) testing of sampling system on a mockup of reactor internals and accurate sample line layout.

The development of a fission product detection system can be based on the experience gained from existing LMFBRs (e.g. EBR-II and Fermi) and, particularly, from the FEDAL system for FFTF. In this instance, the plant tests are part of the FFTF project. Subsequent system modifications will be tested on the Demonstration Plants.

Task 6.1.H - Leakage Detection Instrumentation Development

The purpose of this task is to develop methods of detecting leaks in the sodium boundaries at water or containment atmosphere interfaces. The emphasis for both situations is on response time rather than accuracy.

Subtask 6.1.H.a - Water to Sodium Leakage Detection

The objective is to detect a water to sodium leak as rapidly as possible. Existing methods are either slow or alleviate the effects rather than detect the leak (bursting discs). Failure to detect a leak is unlikely in view of existing methods, but a satisfactory response time may not be attained. The consequences of failure are: a) caustic substances in the secondary sodium circuit requiring a clean-up process, b) rupture of bursting discs requiring clean-up of containment, c) possible damage propagating in steam generator, and d) down-time of loop may exceed that without leakage detection. The studies required include: a) investigation of leakage mechanism and the history of the hydrogen liberated by the reaction, b) study into detection of the leak rather than its consequences (e.g. microphones) c) measurement of hydrogen, oxygen in sodium, and hydrogen in the cover gas, and d) estimates of pressure pulses and development of suitable detectors.

The design of water to sodium leak detection equipment can be based on National Laboratory and industrial tests such as those using the sodium-water loop at Westinghouse Advanced Reactors Division.

Subtask 6.1.H.b - Sodium Boundary Leak Detection

The goal is to detect the leakage of sodium from the circuit boundaries. The flame test can be used for leakage to containment atmosphere; but the circuits will be covered with insulation; and no methods currently exist for leakage detection under these circumstances. Gross failure

can be detected by a fall in circuit inventory as shown by level indicators, but total inventory is difficult to monitor due to liquid expansion with temperature. Work is required to determine methods of detection of sodium in insulation and to develop instruments based on these methods. Failure to develop suitable detects may eventually result in a large sodium inventory outside the circuits which would be a maintenance hazard. Small leaks would not be found until catastrophic failure had occurred.

Sodium boundary leak detectors can be developed by the reactor designers to suit their plant configuration. They can be tested using sodium loops such as GPL-2 described in the Appendix. These units will be proof tested in the FFTF and Demonstration Plants.

Task 6.1.I - Signal Transmission Components Development

The objective is to develop components for the transmission of electrical and hydraulic signals from the sensor to the counter or monitoring point. Many of the instruments require electrical cables or hydraulic lines which share common development problems. These problems can be solved simultaneously as one task.

Subtask 6.1.I.a - Electrical Connectors

The purpose of this subtask is to develop methods and equipment to connect electrical cables so that signal attenuation is minimal and there is no reduction in sensitivity due to cross-coupling leakage, thermal e.m.f's, and noise. At present, there is no capability for under-sodium connectors; those designed for gaseous environments are cumbersome and complex. Failure to develop this equipment may be overcome by design, replacement, and maintenance procedures for some economic penalty.

The work required to accomplish this goal includes: a) a feasibility study for under-sodium connectors including an analysis of the economic incentives for a development program, b) for connectors in the cover gas, existing design data requires unifying and a comparison of uncooled ceramic insulators with cooled organic materials, and c) the design and

testing of prototypes where construction incorporates the most promising methods from b).

The development of cable connectors can be undertaken by companies with experience in this field. They can be tested in the environmental facilities of the National Laboratories.

Subtask 6.1.I.b - Electrical Cables

The objective of this subtask is to develop electrical cables for use in an LMFBR. The types of cables required are low leakage coaxial cables for nuclear instruments and multi-conductor cables for electrical devices and low-impedance signal devices. Although it is claimed that high-temperature cable exists, none is rated for sodium conditions. Failure to develop suitable cabling would preclude the use of in-vessel instrumentation. The studies required include: a) materials studies for cladding, insulation, sealants, and conductors which are chemically compatible and have similar expansion coefficients, b) cable designs using the above materials should be manufactured and tested under in-vessel conditions and tested for mechanical robustness under maintenance conditions (e.g. for drum winding and storage), and c) methods for multiplexing should be studied in order to reduce the amount of in-vessel cabling and number of penetrations.

Suitable electrical cable development can be accomplished in close association with or be undertaken solely by the developers of the instruments and electronic circuitry. Bench and sodium environment tests in loops (e.g. MLTs) are required, but finally the instrument, cable, and circuits require testing together under plant conditions at National Laboratory facilities.

Subtask 6.1.I.c - Transmission Capillaries and Sample Lines

The object of this subtask is to develop design and manufacturing methods, and operating and maintenance procedures to ensure that capillaries and lines remain operable throughout reactor life, and, if necessary, to develop replacement methods. The absolute pressure devices use NaK capillaries to transmit a hydraulic signal. Experience with existing plants

indicates that they are prone to irreversible blockage. Failure in this subtask may require regular replacement of these components. The initial work required is an analysis of the causes of failure in existing plants and rigs and, subsequently, the development of methods to prevent blockage and methods of line clearance and replacement.

The development of procedures for the successful operation and maintenance of sample lines and instrument capillaries can be carried out in association with out-of-pile test loops using carefully controlled impurity levels and test procedures.

6.1.4 Relation to LMFBR Program Plan

The AEC approach to instrumentation development, as described in Volume 4 of the LMFBR Program Plan, is in general agreement with the plan presented here. The AEC plan for each type of instrument can be summarized as follows:

- a. Feasibility studies, where required, of possible methods of measuring a particular variable.
- b. Investigations of instrument materials and components to establish a firm design basis.
- c. Design and manufacture of instruments based on the methods and design bases determined above.
- d. Testing of prototype instruments in statistically significant numbers.
- e. Repetition of c. and d., above, until suitable instruments have been developed.

The Westinghouse plan has been restricted to instruments required in the reference plant. The only area of minor divergence of views is with regard to fission product detection. Westinghouse agrees with the approach to instrument development, but thinks that some emphasis should also be given to the sampling systems. The system selected and used will affect the signal-to-noise ratio and, hence, the overall sensitivity. A study is required to match the characteristics of the sampling systems to the detectors in order to

determine the optimum combinations. This is particularly important in a vented fuel system.

6.1.5 Major Test Facility Requirements

The functional requirements for the test facilities are to reproduce the environments in which the instruments will operate. For devices requiring considerable development, the major requirements are:

- a. Liquid sodium at temperatures up to 1200°F.
- b. Dose rate of 10^6 - 10^7 R/hr.
- c. Ability to simulate thermal shock conditions.
- d. Pressure in the range 15 to 200 psi.

Other facilities include:

- a. Simulation of water to sodium leakage.
- b. Simulation of sodium leakage to gas and insulation.
- c. Rig for testing flow meters up to 4 feet in diameter.

In general, these requirements can be met by using EBR-II, Fermi, and FFTF although the facilities available may be insufficient for the volume of testing required. If this is so, special facilities may be needed.

These new facilities should provide the environments for:

- a. Out-of-core instrumentation.
- b. Standardization and calibration of instruments.
- c. Fission product and failed fuel detection.

6.1.6 Program Schedule

The schedule for the instrumentation development program must be such as to provide instruments to measure all plant parameters in the Demonstration Plant, although development will continue after this time. The time table is presented in Figure 6.1-1, major milestones being defined below. Parts of all of these tasks have been initiated either at the National Laboratories, by industry, or as part of the FFTF program; therefore, no task initiation dates will be shown.

Task 6.1.A - Nuclear Instrumentation

1. Suitable uncooled ionization chamber tested.
2. Completion of testing of high temperature fission chambers.
3. Sodium immersible fission chambers developed. Complete electronics package which satisfies all standards and safety criteria available.
4. In-core fission chambers developed.
5. Operational tests complete and instrumentation proved.

Task 6.1.B - Temperature Instrumentation

1. Report on sonic methods of temperature measurement.
2. Instrument norms established and thimbles designed.
3. Radiation tests on thermocouples and resistance thermometers completed.
4. Thermocouple siting tests complete.
5. Operational tests complete.

Task 6.1.C - Coolant Flow Instrumentation

1. Completion of initial development of transit analysis of temperature fluctuations.
2. Completion of development for ultra-sonic large velocity-head and electromagnetic flowmeters.
3. Completion of operational comparisons of different flow meter.

Task 6.1.D - Pressure Instrumentation

1. Operational data for pipe vessel instruments available. Studies for miniaturization, materials, and replacement techniques for in- and near-core devices complete.
2. In-core and near-core NaK sensors for use at 1400°F available.

Task 6.1.E - Strain Measurement

1. Evaluation of weldable gauges with integral leads and of a method of isolating gauges from sodium complete.
2. Evaluation of flame-spray foil gauges complete.

3. Evaluation of neutron-tolerant materials complete.
4. Evaluation of 1200°F, immersed, neutron-tolerant gauges complete.
5. Extension of temperature range to 1400°F complete.
6. Operational tests complete.

Task 6.1.F - Level Measurement

1. Start testing and calibration of commercial prototypes and models.
2. Operational tests complete.

Task 6.1.G - Fission Product Detection

1. Completion of delayed neutron monitor studies and improvements to EBR-II devices.
2. Completion of cover gas monitor tests (for FFTF).
3. Completion of evaluation of operating characteristics.

Task 6.1.H - Leakage Detection

1. Start testing and calibration of commercial prototypes and models.
First state-of-the-art sodium-to-gas leakage report.
2. Completion of water side study.
3. Completion of tests of commercial cover gas detection.
4. Completion of acoustic detection study.

Task 6.1.I - Signal Transmission Components

1. Cable requirements specified.
2. Start testing of commercial prototypes and models of connector.
3. Thermocouple and neutron detector cable specifications for FFTF completed.
4. Study of replacement techniques complete.
5. Completion of Operational Testing phase.

Figure 6.1-1. Instrumentation Development Program Schedule

6.2 Control Systems Development

6.2.1 Objective

To develop control system technology to a level where the reference 1000 MWe LMFBR plant can be controlled without undue technical and economic risk and without risk to the health and safety of the public.

6.2.2 Program Summary

The program is divided into tasks associated with the main control system in the plant. Unlike the instrumentation program, there is considerable feedback from one control system to another since the plant is not controlled as separate components but as a coherent unit. Several basic areas, however, can be isolated and considered separately.

The four tasks comprising this program are listed below and are described in the next section.

- Reactor Control System Development
- Plant Control Systems Development
- Digital Control and Data Logging System Development
- Reactor Safety System Development

6.2.3 Task Descriptions

Task 6.2.A - Reactor Control System Development

The objective of this task is to develop a system capable of controlling reactivity and reactor power under all circumstances. This task is divided into two subtasks, one associated with the rod drive mechanism and one with the circuitry. As input to this task, basic data concerning reactivity coefficients at all power levels is required.

Subtask 6.2.A.a - Control Rod Drive Mechanism

The objective of this subtask is to develop current rod drive mechanisms into units suitable for controlling an LMFBR. Since the FBR core is much smaller than those of thermal reactors, the rod drive mechanism must be

more sensitive and accurate than that controlling thermal reactors. For mechanisms which move the rod in step increments, this requires redesign of the coils to permit a smaller step size and operation in a more crowded location.

The work required includes: a) redesign of thermal reactor mechanisms for use on an LMFBR, b) studies on the effects of operating in a noble gas environment with sodium vapor present, and c) accuracy and reliability testing under operating conditions.

The adoption of thermal nuclear plant CRDMs to LMFBR use can be carried out by the reactor vendors and tested using out-of-pile loops (e.g. GPL-1).

Subtask 6.2.A.b - CRDM Electronic Circuitry

The goal of this subtask is to develop electronic circuits to operate the control rod drive. The units comprising the circuits include: a) system master pulser, b) module master cyclor and power supply, c) module rod programming and operating logic unit, d) module rod drive selection and hold system control operating circuits, and e) control rod position sensors. The function of the system is to maintain all CRDM's in a state of operability, provide a continuous series of synchronous pulse signals, determine the control rod motions required from plant monitor signals, and send a command signal to the appropriate CRDM's.

The studies to be included are: a) development of the logic circuits, mentioned above, with particular reference to the requirements of multi-module cores, b) development of safety features, and c) investigation of limit cycle stability problems associated with coefficients, deadbands, signal delays, etc.

The design of the electronic circuits required for the control of the CRDM can be executed by the vendors and the testing performed in association with the mechanism tests.

Task 6.2.B - Plant Control Systems Development

The purpose of this task is to develop control systems for application to components in an LMFBR. The major systems, excluding the reactor and turbine-generator, are the pumps, IHX's, and steam generators. There are feedbacks between these systems which will require evaluation. Experience from thermal reactors, e.g., PWR, AGR, HWR, is available for two coolant, multiple loop systems, but the experience requires extension to three coolant plants through simulation methods and application to existing LMFBR's. Although not mentioned specifically, the development, design, and testing of the appropriate circuits is included in these sub-tasks.

All simulation techniques and control logic required in this task can be developed by the reactor designers using digital, analog, and hybrid methods. Comparison with models created by other designers and the National Laboratories is recommended. The prototype circuits and components will be bench-tested by the circuit manufacturer and designer using analog or hybrid simulation of the plant items prior to installation and site testing in conjunction with the actual plant hardware in the Demonstration Plant.

Subtask 6.2.B.a - Sodium Pump Speed Control Systems

This subtask will consider both primary and secondary pump control. The objective is to control systems for the pumps which will: a) control the coolant flows during full and partial power operation so that temperature differences in plant components will not cause stresses greater than design stress limits, b) allow individual and ganged operation of all pumps, and c) allow rapid flow variation so that large temperature rates do not cause reactivity accidents.

The studies require: a) the development of methods of simulation suitable for multi-loop analysis and experimental verification where possible, b) development of control circuitry for a multiple loop, three coolant heat removal system including the interconnection and ganged control of the pumps and feedback logic from the main plant components,

c) investigation of the sensitivity of the control loops to changes in system variables, and d) definition of operational procedures for individual pumps and ganged pumps under normal and fault conditions. This last job requires input data regarding forbidden operational areas for system variables.

Subtask 6.2.B.b - Steam Generator Control Systems

The goal of this subtask is to develop a control system for the steam generator units. The system must be capable of: a) providing feedback to the secondary and primary pump speed controls and reactor control system, b) controlling the feedwater flow so that the load demand is met and the steam conditions are within design limits for full and partial power operation, c) controlling the feedwater temperature at full and partial power to match the sodium temperature and maintain material stresses within design limits, d) locating the position of the boiling boundary and ensuring that boiler dryout and turbine flooding do not occur, and e) isolating the steam generator on detection of a gross leak and providing a trip signal to the reactor.

This demands considerable development of logic and circuitry, and feedback between the steam circuit and control system designers. The studies required include: a) development of adequate steam generator simulation models and their verification, b) development of parallel channel flow stability models to predict flow oscillations, c) development of security of control to prevent loss of feedwater flow and definition of the minimum and maximum flow rates required to prevent boiler dryout and turbine flooding under various conditions, d) determination of the accuracy required from temperature and pressure regulation and the effects of bandwidth on stability, e) determination of the effects of thermal transients due to loss of feedwater flow, f) an investigation into the detection of leaks and sodium-water reactions and the total isolation of the steam generator from the secondary and tertiary systems, and g) a study into the feedback from one generator to the other through the steam system.

Task 6.2.C - Digital Control and Data Logging System Development

The objective of this task is to develop a digital system capable of full-filling control, data logging, and fuel management functions in an LMFBR. The digital hardware for this task will be a commercial process computer; however, the interface between the monitors and computer may require some development, although modified hybrid computer equipment should suffice. This reduces the task to the development of computer software. The system should be able to: a) monitor plant sensor information, calculate significant plant variables, compare with established performance criteria, and determine and execute the control trajectory required to maintain safe plant operation, b) collect and log received sensor data and calculated parameters as required by the operator and plant operating procedures, and c) using data received from in-core sensors and activation monitors, maintain an inventory of fissile materials in the plant and recommend action for breeding optimization (see section on Computer Assisted Fuel Management System).

The software requiring development includes programs for: a) data acceptance, manipulation, and logging, b) alarm recognition and analysis, c) maintenance of fissile inventory and fuel management, and d) plant supervision and control. Some hardware development may be required to: a) improve reliability and, hence, plant security and safety, b) switch to reserve equipment if failure should occur, and c) improve analog/digital signal conversion speed.

The computer hardware development will be carried out by the computer manufacturers (e.g. Westinghouse Computer Services Division). The development of the required software can be performed by the computer vendor in association with the reactor designer. Some parts of the software package (e.g. data handling) may be prototype-tested on other reactors (FFTF or Demonstration Plant), but laboratory testing will predominate until reactor testing commences.

Task 6.2.D - Reactor Safety System Development

The objective of this task is to develop a system to prevent the plant exceeding its established design limits during normal or abnormal operation. Although the basic logic and circuitry has been developed for thermal power reactors, considerable development is required in both fields due to the increase in the number of signals to be handled and the more rapid rate of instrument failure due to temperature and irradiation.

The work required includes: a) development of digital, analog, and hybrid models and the identification of the significant transients for use as the trip system design basis, b) evaluation of safety channel performance and operating characteristics for the trip system for the design basis transients, c) development of the required sensor circuits and logic to fulfill its safety function including due allowance for sensor drift and failure, and d) identification of any further development required.

The reactor designer can develop the simulations and safety logic required to design the Reactor Safety Systems. This part of the program, including the circuit testing, will be similar to that for the Reactor Control System.

6.2.4 Relation to LMFBR Program Plan

There is good agreement between the AEC Program Plan and the Westinghouse approach regarding control and computer aspects of LMFBR's. The AEC plan involves:

- a. Developing and applying simulation methods to LMFBR's and associated plant.
- b. Determining control requirements and methods of satisfying these requirements.
- c. Verification of these methods by theory and experiment.
- d. Developing and applying modern control methods.

- e. Applying computers as widely and usefully as possible to all aspects of reactor operation.

The Westinghouse plan has identified particular areas which will require special attention, but no basic difference of opinion exists in this area.

6.2.5 Major Test Facilities Required

No major facilities are required that cannot be provided by: a) a well-equipped computer laboratory, such as the WARD Hybrid Computer Facility which is described in the Appendix, and b) a computer and interface on site with an existing LMFBR, SEFOR, FFTF, or a Demonstration Plant.

6.2.6 Program Schedule

The schedule for the development of station control logic and hardware is shown in Figure 6.2-1. The major milestones are defined below. These tasks have been initiated and, therefore, no initiating milestones are shown.

Task 6.2.A - Reactor Control System Development

1. State-of-the-art report complete.
2. First plant control studies complete.
3. Completion of control investigation in plant and components. Reactor control methods optimized. First results on advanced control methods.
4. Improved simulation.
5. Second-round results.

Task 6.2.B - Plant Control Systems Development

1. First control studies of individual plant-items complete.
2. Complete simulation of plant.
3. Improved plant simulation.
4. Verification of design methods against observed performance.

Task 6.2.C - Digital Control and Data Logging System Development

1. Completion of computer applications study.

2. Initial results concerning the stability of digitally controlled plants in a closed-loop mode.
3. Computer applied to FFTF.
4. Computer control language or substitute available.
5. Prototype control displays.

Task 6.2.D - Reactor Safety Systems Development

1. Complete equipment definition.
2. Complete initial equipment tests.
3. Complete revision of equipment specifications.

Figure 6.2-1. Control Systems Development Program Schedule

7. SODIUM TECHNOLOGY

7.1 Liquid Metal Purification Development

7.1.1 Objective

To develop reliable, economical systems and components which will maintain the sodium and NaK purity required for satisfactory long-term operation of the 1000 MWe LMFBR. The principal impurities which will be controlled are nitrogen, oxygen, carbon, and hydrogen, any one of which can react with system components to cause premature failure.

7.1.2 Program Summary

The program is divided into four tasks, each describing the development of a purification method starting with state-of-the-art reviews and tests to obtain basic data, and then progressing, through small scale tests, to prototype tests on the Demonstration Plant. These four tasks described in the following section, are listed below:

- Cold trap development
- Hot trap development
- Mechanical separation methods development
- Distillation development

7.1.3 Task Descriptions

Task 7.1.A - Cold Trap Development

The objective of this task is to develop a cold trap which will provide reliable and efficient removal of oxygen and hydrogen from the liquid sodium and NaK coolants. Investigation by calculation and experiment will be conducted to obtain necessary basic information on the precipitation of impurities from liquid sodium. Significant variables for investigation will include temperature, time, velocity, size and density of precipitate, heat transfer, nucleation sites, and effect of trace impurities and fission products such as cesium. Based on this work, several conceptual designs of cold traps will be developed. Candidate

designs which can provide maximum useful design information will be selected for testing in a liquid sodium loop. Tests at various contaminant levels will include measurement of contaminant concentrations, pressure drop, throughput, temperature and impurity distribution, and maximum capacity. Regeneration methods versus replacement of trap will be considered. System control will be analyzed to evaluate response to transients in flow, temperature, and contaminant levels. An optimized design at 1/4 to full scale size will be tested extensively in an experimental sodium loop and hopefully in the FFTF in the system of the in-core loops. On the basis of test results, a full scale cold trap will be designed for installation in the Demonstration Plant.

Task 7.1.B - Hot Trap Development

The objective of this task is to develop reliable and efficient traps for the removal of small quantities of oxygen and carbon. The oxygen removal trap will be required if oxygen levels are desired below the 10 ppm concentration which is the approximate lower limit for cold traps. The carbon removal traps are required to maintain a carbon concentration in the liquid sodium coolant which will prevent excessive removal or addition of carbon to critical components such as fuel cladding. Several conceptual designs will be prepared for each process, utilizing the information now available on metal types and forms for gettering, characteristics of various oxide layers, temperature requirements, residence time, and filtration requirements. The more promising candidate designs will be tested in a small scale sodium loop. Following this, an optimized design at 1/4 to full-scale size will be given extensive tests in a larger sodium loop. Results of these tests will provide design information for the design of hot traps for installation in the FFTF in-core loops and the Demonstration Plant.

Task 7.1.C - Mechanical Separation Methods

The objective of this task is to evaluate the feasibility and reliability of purifying liquid sodium by mechanical separation of precipitated impurities. Conceptual designs will be prepared for filters, settling

basins, and centrifuges. These concepts will be related to the design of precipitate-type equipment, such as cold traps and hot traps, and may take the form of a variation in design of one of these. The more promising concepts will be tested in the laboratory using a liquid sodium loop with varying amounts and types of precipitates. The test program would include measurement of removal efficiency, maintenance problems, probable life of equipment, and methods for removal of concentrated precipitates from the liquid sodium system. Prior to final concept selection, components of 1/4-scale to full scale will be designed and thoroughly tested in a pilot plant. Based on these test results, a selected concept would be used to design a full scale unit for test in the Demonstration Plant.

Task 7.1.D - Distillation Development

The objective of this task is to evaluate the sodium distillation process as a satisfactory and reliable method for sodium purification. Bench scale stills will be designed and operated with necessary instrumentation to determine the analysis of distillate and residue. The effect of distillation rate, materials of construction, temperature, and pressure will be evaluated in these tests. The advantage of rectification trays added to the still will be analyzed for possible bench scale evaluation.

If bench scale tests provide satisfactory positive results, components will be designed and tested as part of a liquid sodium loop. Tests will include evaluation of batch versus continuous distillation, effect of transient loop conditions, methods for impurity removal, and range of impurity level which can be handled. Upon completion of these tests, a full scale distillation unit will be designed for testing in the Demonstration Plant.

7.1.4 Relationship to LMFBR Program Plan

The scope of the proposed program is in general agreement with Task Area 5-2 "Sodium Purification" in Volume 5 of the LMFBR Program Plan. The time scale for the development, as proposed in this program, also is in agreement with that described in the LMFBR Program Plan.

7.1.5 Major Test Facility Requirements

The following test facilities will be required in addition to the general sodium laboratory facilities normally available.

Large Sodium Test Facility (Tasks 7.1.A, 7.1.B, and 7.1.C)

Small bends scale loops will be required for obtaining and evaluating basic information for the initial screening test series.

A large sodium test facility will be required consisting of a liquid sodium loop which contains large amounts of sodium and to which it would be convenient to add purification equipment with means for injecting measured quantities of impurities. The maximum circulation rate in the purification train would be approximately 100 gpm. A holdup tank would be required to improve mixing and sampling to provide generally stable operating conditions. Two systems of this type existing at Westinghouse ARD are the General Purpose Loops 1 and 2 as described in the Appendix.

Facilities for Task 7.1.D

- a. For bench scale development, no major facilities will be required in addition to the components under test.
- b. For pilot plant tests, a liquid sodium loop would be required. Maximum circulation rate would be approximately 100 gpm.

7.1.6 Program Schedule

The timetable for this program is presented in Figure 7.1-1. Major milestones indicated on the time table are defined as follows:

Task 7.1.A - Cold Trap Development

Task 7.1.B - Hot Trap Development

Task 7.1.C - Mechanical Separation Methods

1. Start program and facility definition.
2. Facility complete; initial tests started.
3. Initial tests completed.
4. Complete design revisions. Start second series of tests.
5. Complete tests. Start design of unit for FFTF in-core loop.

6. Complete design for Demonstration Plant.

Task 7.1.D - Distillation Development

1. Bench scale program and test facility defined.
2. Test facility installation completed.
3. Test series completed.
4. Pilot plant program and facility defined.
5. Pilot plant facility installation completed.
6. Test program completed.
7. Complete design for Demonstration Plant.

TASK	TITLE	CALENDAR YEAR						
		1969	1970	1971	1972	1973	1974	
7.1.A	DEVELOP DESIGN OF COLD TRAP	<div>1</div>	<div>2</div>	<div>3</div>	<div>4</div>	<div>5</div>	<div>6</div>	
7.1.B	DEVELOP DESIGN OF HOT TRAP	<div>1</div>	<div>2</div>	<div>3</div>	<div>4</div>	<div>5</div>	<div>6</div>	
7.1.C	DEVELOP MECHANICAL SEPARATION METHODS	<div>1</div>	<div>2</div>	<div>3</div>	<div>4</div>	<div>5</div>	<div>6</div>	
7.1.D	DEVELOP DISTILLATION METHOD	<div>1</div>	<div>2</div>	<div>3</div>	<div>4</div>	<div>5</div>	<div>6</div>	<div>7</div>

Figure 7.1-1. Liquid Metal Purification Development Program Schedule

7.2 Cover and Hot Cell Gas Purification Development

7.2.1 Objective

To develop reliable, economical systems and components which will maintain cover gas of required purity and pressure at all liquid sodium-gas interfaces in the reference 1000 MWe LMFBR plant. The basic functional requirements of the systems include: a) prevention of sodium fire, b) removal of radioactive gaseous fission products as they accumulate on release from the liquid sodium surface, c) prevention of physical or chemical actions at the sodium-gas interface which might result in formation of harmful impurities in the liquid sodium coolant, and d) control of system sodium pressure.

7.2.2 Program Summary

The program is divided in three tasks, each covering a separate junction of the inert gas purification system. These functions are listed below; tasks required to develop components to perform the function are described in the next section.

- Remove krypton and xenon from the cover gas
- Remove non-radioactive gaseous impurities from the cover gas
- Prevent sodium and sodium oxide contaminants from contaminating the System.

7.2.3 Task Descriptions

Task 7.2.A - Develop Adsorbents for Removal of Krypton and Xenon from Cover Gas

The objective of this task is to provide design information for reliable full scale adsorption systems. Activated charcoals and other candidate adsorbents, such as alumina and silica gel, will be tested in systems designed to remove krypton and xenon from argon, helium, or nitrogen. Molecular sieves which provide the additional feature of controlled pore size will be included as candidate adsorbents. Fundamental tests will be conducted to obtain basic data on adsorption of trace impurities

by various adsorbents. Bench scale experiments will provide information on quantities of adsorbent required, geometry of beds, optimum temperature and pressure conditions, refrigeration requirements, reactivation conditions, expected adsorbent life, and related parameters. Fission gas concentrations will be varied through a wide range to evaluate factors which control the adsorption rate. Decay heat will be simulated in the beds to determine temperature distribution in the bed and its effect on adsorbent capacity.

Following completion of "bench scale" tests, pilot plant size components will be designed and tested. Emphasis will be on repeated operations and reliability of components operating under design conditions. When adequate design information has been acquired, a prototype unit will be designed for installation in the Demonstration Plant.

Task 7.2.B - Develop Methods for Removal of Non-Radioactive Gaseous Impurities from Cover Gas

The objective of this task is to develop simple, reliable systems for purification of cover gas supply and recirculation streams. Maximum permissible concentrations will be established for impurities such as oxygen, hydrogen, nitrogen, water, carbon monoxide, and carbon dioxide. Several purification systems will be evaluated and the more promising ones will be tested in the laboratory. These will include NaK bubbler, low temperature distillation, chemical reaction (catalytic and non-catalytic), and separation by membranes. For the preferred system, a prototype design will be prepared to permit testing in the Demonstration Plant.

Taks 7.2.C - Develop Sodium Vapor and Aerosol Traps

The objective of this taks is to develop simple, reliable component designs to prevent contamination of cover gas system components by sodium and sodium oxide. Experimental work will be done as required to obtain basic data on sodium vapor condensation and those factors which control the formation of sodium aerosols. Several conceptual designs for traps will be developed and evaluated. The more promising designs will be

tested for efficiency, regeneration methods, reliability, and overall life. Concepts will include scrubbing, condensing, freezing, adsorption, centrifugal separation and filtration. The experimental facilities will be planned to support the investigation of conditions which contribute to the formation of aerosols. Concepts for suppression of aerosol formation over liquid pools will be formulated and evaluated. After small scale tests have been completed and a preferred design is selected, a full scale system will be built and operated as a pilot plant. A prototype system and components will then be designed for installation and test in the Demonstration Plant.

7.2.4 Relationship to LMFBR Program Plan

The scope of the proposed program is in general agreement with the LMFBR Program Plan. Task Area 5-7, "Cover Gas and Maintenance Atmospheres," covers portions of all three tasks of this program. In addition, Tasks 5-3.3, "Fission Product Removal Studies," and 5-3.6, "Fission Product Analytical Method for Cover Gas," are directly applicable to Task 7.1-A of this program.

7.2.5 Major Test Facility Requirements

Prior to the Demonstration Plant Operation, the following test facilities will be required in addition to the general sodium laboratory facilities normally available.

Develop Absorbents for Removal of Krypton and Xenon from Cover Gas (Task 7.2-A)

- a. Bench scale facility including gas supply and mixing system, gas circulation system, liquid nitrogen supply system, and analytical equipment. Initial tests will be on a system of approximately 1/500 scale relative to the 1000 MWe plant.
- b. Pilot plant facility including gas supply and mixing system, and gas circulation system. The liquid nitrogen supply and analytical equipment from the bench scale facility can be used in this facility. Gas circulation will be at rates of 1/10 or larger of the expected design rates for the 1000 MWe plant.

Develop Sodium Vapor and Aerosol Traps (Task 7.2-C)

- a. Bench scale facility including a liquid sodium loop with liquid-gas surfaces, a gas circulation system with instrumentation and control, and miscellaneous heat exchangers. Initial tests will probably require circulation rates of 1/20 scale relative to the 1000 MWe plant.
- b. Pilot plant facility including a liquid sodium loop with liquid-gas surfaces of various sizes and geometry, a gas circulation system with instrumentation and control equipment, and miscellaneous heat exchangers. Gas circulation rates will be at approximately 1/2 scale relative to the 1000 MWe plant.

Develop Methods for Removal of Nonradioactive Gaseous Impurities from Cover Gas (Task 7.2-B)

- a. Bench scale facility including contaminant gas supplies, mixing and circulating system, and gas analysis equipment. Initial tests will probably require circulation rates of 1/100 scale relative to the 1000 MWe plant.
- b. Pilot plant facility including contaminant gas supplies, mixing and circulating system, and gas analysis equipment. Gas circulation will be at rates of 1/10 or larger of the expected design rates for the 1000 MWe plant.

7.2.6 Program Schedule

The timetable for this program is presented in Figure 7.2-1. Major milestones indicated on the timetable are defined as follows:

Task 7.2.A - Develop Adsorbents for Removal of Krypton and Xenon from Cover Gas

Task 7.2.B - Develop Methods for Removal of Non-Radioactive Gaseous Impurities from Cover Gas

Task 7.2.C - Develop Sodium Vapor and Aerosol Traps

1. Bench scale program and test facility defined.
2. Test facility installation completed.

Figure 7.2-1. Hot Cell Gas Purification Development Program Schedule

3. Test series completed.
4. Pilot plant program and facility defined.
5. Pilot plant facility installation completed.
6. Test program completed.
7. Design completed for Demonstration Plant.

7.3 Development of Non-Gaseous Fission Product and Active Corrosion Product Management Schemes

7.3.1 Objective

To develop an understanding of fission product and active corrosion product identity and behavior in sodium and cover gas (excepting gaseous fission products). From this knowledge, it will be necessary to develop an adequate fission product management scheme which will meet reference plant requirements.

7.3.2 Program Summary

The program on Fission Product and Active Corrosion Product Management covers:

- a. Analytical studies aimed at estimating the magnitude of the problem, pinpointing specific problem areas, and establishing activity management system design bases.
- b. Experimental studies ultimately leading to the specification of a satisfactory radioactivity management scheme for the LMFBR coolant system. These studies cover laboratory and loop investigations of the physical and inorganic chemistry and transport behavior of fission products and active corrosion products in liquid sodium and, where applicable, argon gas systems.
- c. Based on the foregoing work, and a follow up of fuel irradiation studies being carried out under other LMFBR programs, the need or desirability of a system proof test by means of an in-pile loop will be evaluated, and, assuming there is an affirmative decision, the required facility and test program will be implemented.
- d. An additional task included in this program is to study and develop methods of decontaminating removable components and reactor structural sections of fission and active corrosion products.

This program described in the next sections can be broken into five tasks listed below:

- Theoretical and Analytical Studies of Fission Products in Sodium
- Experimental Studies of Fission Product Behavior in Sodium Systems
- Study of Fission Product Transport Across Sodium/Gas Interface
- Definition of In-Pile Loop Experiments
- Development of Fission Product Decontamination Methods

7.3.3 Task Descriptions

Task 7.3.A - Theoretical and Analytical Studies of Fission Products in Sodium

A quantitative estimate of the nature and magnitude of the radioactivity problem in LMFBRs will be made by:

- a. Estimating the fission product source rate to the coolant from defected and/or vented fuel based on experimental data; active corrosion product sources will be based on general corrosion data and nuclear activation calculations.
- b. Thermodynamic and kinetic analyses of expected activity distribution throughout the reactor system. Chemical interaction of the active products with sodium and its contaminants, solubility and physical state of the products, deposition or plateout behavior, and purification system effects will be considered.
- c. Estimating the maximum allowable limits of fission and corrosion product activities in the coolant and deposited on system surfaces as imposed by maintenance, operation, and safety requirements.

Task 7.3.B - Experimental Studies of Fission Product Behavior in Sodium Systems

The objective of this task is to study, experimentally, the chemical behavior of fission products in sodium to determine the species and mechanisms involved in: a) fission product interactions with sodium, cover gas, and containment materials, and b) fission product removal processes.

Initially, basic chemistry studies will be performed to determine the

nature of dissolved fission products in sodium, and the interactions of these dissolved species with impurities, sodium cover gas, and containment materials. In particular, fission product adsorption isotherms on a variety of structural and potential getter materials will be measured in essentially quiescent sodium systems.

The phenomenon of fission product plateout and removal will be studied in small dynamic sodium loops consisting of 1/2-inch to 1-inch pipe and operating with flow rates in the 10 - 50 gpm range. These experiments will be of a scoping nature in that they will examine cold trapping, carbon trapping, and hot trapping as possible means of fission product removal. Distillation also will be evaluated, and if deemed feasible, a scoping experiment will be performed.

Other fission product removal techniques which appear technically feasible will be experimentally evaluated in the laboratory-scale sodium loops. The most promising of the fission product removal techniques will then be evaluated on a pilot plant scale to establish design and economic bases for large LMFBR plant application.

Task 7.3.C - Study of Fission Product Transport Across Sodium/Gas Interface

The objective of this task is to determine the extent of fission product transport across a sodium/gas interface by evaporation and/or entrainment, and their subsequent behavior in the vapor phase.

A test apparatus containing a sodium pool exposed to a large gas environment must be designed and constructed. Mass transfer rates for fission products and sodium vapor will be determined as a function of gas and sodium temperatures, sodium pool diameter, and purity of gas.

Rates of fission product sodium deposition on geometric surfaces placed within the convective streams will be studied to determine the extent to which fuel handling equipment may become contaminated during operation over sodium pools.

Various techniques and equipment for reducing mass transport of fission products and sodium will be investigated. These will include gas curtains, iris shutters, and vacuum sweepers.

Task 7.3.D - Definition of In-Pile Loop Experiments

The objective of this task is to define the requirements for an in-pile loop experimental verification of the fission product management concept.

Uncertainties as to the performance of fission product management concepts in a reactor environment may necessitate verification of these concepts in an in-pile loop experiment before final design of Demonstration Plant equipment. This task will assess the requirements for such an experiment, provide a preliminary definition of the experiment, and will evaluate the capabilities of existing facilities for performing the experiment. A recommendation for the complete in-pile program will result.

Task 7.3.E - Development of Fission Product Decontamination Methods

The objective of this task is to establish efficient methods of fission product and active corrosion product decontamination of both removable components and reactor structural sections.

The information obtained in the other tasks of this program will serve as input to this task regarding the type, amount, and expected locations of deposited products in the reactor system. Screening tests will be performed of candidate decontamination methods for effectiveness of fission product removal.

Specimens will be contaminated synthetically in fission product study loops, and contaminated pieces from reactors or in-pile loops, as available, will be used. A one-step decontamination process for both sodium and fission products has obvious economic advantages. At present Westinghouse favors jet cleaning which uses steam of high-impact energy and velocity and, for this reason, may be effective for fission products.

The depth of fission product penetration into surface will be determined if it should prove necessary to remove significant amounts of surface material because of fission product diffusion into the structure.

7.3.4 Relation to LMFBR Program Plan and Other Programs

Current programs of work are being conducted at LASL, BNL, AI, and BNWL.

These are directly related to the Westinghouse program, and together they cover virtually the whole AEC suggested requirements for fission products studies in LMFBR Program Plan, Volume 5, Task Area 5-3, "Fission Products in Sodium Systems."

7.3.5 Major Test Facility Requirements

Fission product deposition and purification loops in the 1 to 50 gpm range, plus a larger system (200 gpm) for pilot purification studies, must be available. The potential need for an in-pile loop also must be recognized.

7.3.6 Program Schedule

The program schedule is shown in Figure 7.3-1 with the major milestones listed as follows:

Task 7.3.A - Theoretical and Analytical Studies of Fission Products in Sodium

1. Initiate studies of fission product release rate, thermodynamic behavior and effects on plant operation and maintenance.
2. Complete studies and make recommendations.

Task 7.3.B - Experimental Studies of Fission Product Behavior in Sodium Systems

1. Initiate experimental studies.
2. Complete studies of physical chemistry of fission production in sodium.
3. Complete fission product removal studies.
4. Complete fission product plate-out studies.
5. Complete pilot demonstration of purification processes.

Task 7.3.C - Study of Fission Product Transport Across Na/Gas Interface

1. Initiate studies of transport.
2. Complete studies of sodium evaporation rates.
3. Complete studies of fission product evaporation transport.

Task 7.3.D - Definition of In-Pile Loop Experiments

1. Initiate evaluation of requirements
2. Complete definition of in-pile program

Task 7.3.E - Development of Fission Product Decontamination Methods

1. Initiate fission product decontamination screening tests
2. Complete screening tests
3. Complete depth of fission product penetration studies
4. Complete fission product decontamination studies

Figure 7.3-1. Development of Non-Gaseous Fission Product and Active Corrosion Product Management Schemes

7.4 Sodium Quality Control

7.4.1 Objective

To develop active sodium in-line monitors to provide adequate quality control. To investigate aspects of sodium chemistry of immediate benefit to quality control.

7.4.2 Program Summary

The program consists of four tasks providing for the development of a system for control of sodium quality by in-line metering. Should certain techniques prove inadequate, then alternatives will be available under this scheme. The sodium chemistry study described in program 7.5 of this report will assist interpretation of all in-line meters.

7.4.3 Task Descriptions

Task 7.4.A - Oxygen Meter Development

The on-going Westinghouse program of oxygen meter development should be pursued. This program and programs of the UNC meter should be directed towards producing an improved electrolyte, particularly with respect to impurity content.

The development, therefore, is divided into two phases: a) development of an improved electrolyte of good compatibility with sodium (long life) and of high ionic conduction, and b) construction and assessment of a meter, covering items listed Information Needs.

Demonstration Plant experience will provide relevant data to final assessment stages.

Task 7.4.B - Carbon Meter Assessment^a

This will commence when this meter is commercially available. Exact areas for assessment cannot be specified at present. Demonstration Plant experience will provide relevant data to final assessment stages.

- a. Existing C and H meter development programs are assumed to continue.

Task 7.4.C - Hydrogen Meter Assessment^a

This will commence when the meter is commercially available. Exact areas for assessment cannot be specified at present. Demonstration Plant experience will provide relevant data to final assessment stages.

Task 7.4.D - Continuous Indication Plugging Meter Assessment

It is assumed that the U.K. meter is commercially available. Assessment of items listed in Information Needs will interface with the impurity studies of Program 7.5, Sodium Sampling and Analysis. The aim is to produce a device providing back-up information. Envisaged is a situation in which knowledge of sodium impurity chemistry still will be far from complete. Unequivocal interpretation of the specific in-line meters may not always be possible. In addition, a useful guide to start-up cleaning procedures will be available.

7.4.4 Relation to LMFBR Program Plan

The AEC approach to sodium chemistry quality control in Volume 5, "Sodium Technology," of the LMFBR Program Plan is very broad with virtually all conceivable aspects being covered. There are far too many aspects for one company to approach. The program presented here selects items which are considered immediately relevant to plant operation. The Program Plan devotes much effort to non-specific impurity monitors and in particular the resistivity meter.

7.4.5 Major Test Facility Requirements

A circulating sodium loop will be required for general application to all Tasks. The Westinghouse General Purpose Loops, 1 and 2 described in the Appendix, could be used for this application.

a. Existing C and H meter development programs are assumed to continue.

7.4.6 Program Schedule

The time table for this development program is presented in Figure 7.4-1. Major milestone definitions, as indicated in scheduling figures, are presented below:

Task 7.4.A - Oxygen meter development

1. Improved electrolyte available
2. Meter assesement completed
3. Demonstration Plant meters installed and providing final performance data

Task 7.4.B - Carbon meter assessment

1. Meter assumed available and installed on test facility
2. Meter assessment completed
3. Demonstration Plant meters installed and providing final performance data

Task 7.4.C - Hydrogen meter assessment

1. Meter assumed available and installed on test facility
2. Meter assessment completed
3. Demonstration Plant meters installed and providing final performance data

Task 7.4.D - Continuous-indication Plugging Meter Assessment

1. Meter assumed available and installed on test facility
2. Meter assessment completed

Figure 7.4-1. Sodium Quality Control Program Schedule

7.5 Sodium Sampling and Analysis

7.5.1 Objective

To establish competence in the sampling and analysis of sodium in nonradioactive and radioactive systems to meet the functional requirements of the LMFBR program. The analytical competence includes the determination of concentration and species of the interstitial impurities in the range from 0.1 to 20 ppm.

7.5.2 Program Summary

Any program to develop an adequate methodology for the sampling and analysis of sodium coolants used in the LMFBR program should be aimed primarily at two main points: first, the determination of species as well as concentration for the interstitial impurities; and second, at the provision of adequate standard samples for use in establishing inter-laboratory precision and accuracy. The methods also would have to be sensitive, precise, and accurate as well as being reasonably economical and ideally adaptable to inline application. The specific methods sought would be for oxygen, carbon, nitrogen, hydrogen, metallics, and halide impurities. The end-use of these methods would be their application to support and establish the operation of on-line monitors and to establish the effectiveness of hot traps, cold traps, and other purification devices used in experimental loops.

A secondary, but especially important, part of the program is the preparation of suitable standard samples of sodium containing low concentrations of these impurities in known forms. The actual form of such samples and the storage method selected probably will be dependent on the impurity sought, but, in most cases, a preparation procedure would be required to supply several dozen samples, each sealed in a container providing long shelf life. Purification methods developed in other tasks may require refinement to produce adequate material for such standards.

Finally, attention would be given to the control of contamination during sample collection and storage, and during the analytical processes.

The tasks in this program are listed below and are described in the next section.

- Sampling and Handling of Sodium
- Determination of Oxygen Concentration and Species
- Determination of Carbon Concentration and Species
- Determination of Hydrogen Concentration and Species
- Determination of Nitrogen Concentration and Species
- Determination of Metallic and Halide Impurities
- Sampling Methods for Cold Traps and Hot Traps
- Preparation of Standard Samples of Sodium
- Nondestructive chemical analysis

7.5.3 Task Descriptions

Task 7.5.A - Sampling and Handling of Sodium

This task involves the development of sampling and handling procedures for obtaining representative samples from nonradioactive or radioactive sodium systems and suitable storage procedures. All sampling methods would be evaluated specifically for application to the analytical techniques developed, freedom from segregation within the sample used, and cost and reliability. The latter points suggest small sample sizes which are not possible at the lower levels of interest in this work.

Specific sampling methods for investigation are:

- a. Vacuum distillation sampling for on-line analyses
- b. Bypass sample lines for freeze-out sampling
- c. Other sampling methods, i.e., thief sampling, spigot sampling.

These methods require careful evaluation at this time.

Task 7.5.B - Determination of Oxygen Concentration and Species

The development of specific and accurate methods for the oxygen-containing species present in sodium is the most important single analysis. The species in nonradioactive or radioactive sodium at the 0.1 to 20 ppm level are of great importance. The classical methods, mercury amalgamation

and vacuum distillation, are not specific and are usually only practical down to an apparent oxygen level of about 2 ppm. These methods would be reexamined and carefully evaluated as new methods are sought and applied to this problem. At this time, NaOH is determinable by the mercury amalgam reflux method down to a level of 0.1 ppm and is discussed in Task 7.5.D.

Task 7.5.C - Determination of Carbon Concentration and Species

Specific chemical analyses for carbon species in sodium are not well established, and gross carbon methods are only satisfactory at levels above 2 ppm. Gross segregation occurs in carbon samples. In bypass freeze sample tubes, severe axial gradients are seen in the solidified sodium.

Total carbon determination is effected by dry combustion, and significant blanks hold the detection limit at about one ppm. Carbonate carbon is determined by careful dissolution in water and collection of the off gases for analysis by gas chromatography. Future methods would include:

- a. Dry oxidation - total carbon
- b. Oxidation of hydrolized sodium - species (?)
- c. Gas liberating reactions on sodium - species (?)

The effect of segregation during sampling would be evaluated in detail in all methods tested for carbon since pronounced segregation is seen in total carbon samples. The gas-liberating reactions would be applicable to large molten samples since no manual handling is involved. Such methods could be set up readily for remote use on highly radioactive sodium systems.

Task 7.5.D - Determination of Hydrogen Concentration and Species

The amalgam reflux method would be developed to a sensitivity of 0.1 ± 0.02 ppm for hydroxide form and non-hydrogen and applied to nonradioactive and radioactive sodium systems.

Dissolved hydrogen and hydrocarbons will be sought in sodium, and this work would be continued along with the amalgam reflux technique. The

results would be evaluated and, if appropriate, the vacuum extraction (membrane-diffusion) method would be applied for hydrogen only and developed for remote application such as a radioactive sodium systems. Isotopic dilution techniques would be used extensively in this work to permit additions, or "spiking", of samples with known amounts of deuterium in known chemical form and to allow subsequent analysis by mass spectrometry.

Task 7.5.E - Determination of Nitrogen Concentration and Species

Methods will be examined for determination of total nitrogen, metallic nitrides, cyanides, and amides in sodium. The most promising techniques will be selected for development to achieve limits of detection below 1 ug of nitrogen.

Task 7.5 F - Determination of Metallic and Halide Impurities

Analytical methods will be developed for the metallic and halide impurities found in sodium related to:

- a. Base metal components in the system
- b. Metallic additives in fuel cladding and in hot traps
- c. Impurities usually present in the sodium charge such as alkaline earths and other alkali metals
- d. Fuel leaching into the sodium
- e. Halogens present initially or added as fission products

Emission spectroscopy, atomic absorption spectroscopy, and isotopic dilution methods will be employed in this work. Isotopic dilution methods, using added radioactive species as tracers, will be used to measure recovery elements in separations and provide greatly improved precision and accuracy when samples are not radioactive. Sensitivities of about 1 ppm will be the target for most elements except fissile fuel components such as Pu, U, where ≤ 0.01 ppm would be more appropriate. In systems, such as cold traps or hot traps, where large amounts of impurities have been collected, the scope would be expanded to ensure removal of representative samples of the impurities and the use of X-ray diffraction to characterize

the structure of the collected material. This application also would require the development of sound sampling procedures for large objects and the development of procedures for the separation of sodium and particulate contamination.

Task 7.5.G - Sampling Methods for Cold Traps and Hot Traps

Since the trapping of impurities is one effective means of maintaining a clean sodium system, the analysis of such traps is an important effort in the LMFBR program. Improved designs of cold traps or hot traps specially applicable to rapid sampling should be studied as an effective means of approaching contamination levels in loop experiments and categorizing the actual processes responsible for the removal of contamination in the traps.

Task 7.5.H - Preparation of Standard Samples of Sodium

The preparation and storage of suitable samples of high purity sodium is an important part of this analytical program. Purification methods would be developed and applied on a suitable scale to prepare batches of high purity sodium suitable for use as standard samples. These samples would be used within the laboratory as quality control samples and would be available for inter-laboratory studies to determine the accuracy and precision of various methods.

Task 7.5.I - Non-Destructive Chemical Analysis

Upon completion of destructive chemical analysis, there should be non-destructive methods that would immediately give an indication as to the condition of the sodium as a surveillance technique. This type of analysis (NDT) would be effective for plant operation, and different instruments should be investigated for their adaptability. Among the candidates for (NDT) are gamma and alpha spectrometers, X-Ray fluorescent spectrometers, direct reading optical emission and mass spectrometers, ultrasonic echo and transmission detectors, high-precision freeze-point depression regulators, and magnetic-DC current purifiers. These are but a few of the more popular instruments.

7.5.4 Relation to LMFBR Program Outline

The tasks presented in this section are similar to those described in Task Area 5.5, "Sampling and Analysis - Laboratory Methods" of Volume 5 of the LMFBR Program Plan. The target sensitivities in this section are more realistic and are generally achievable by 1972 or 1973.

7.5.5 Major Test Facility Requirements

Complete implementation of the tasks described in this program would require no major facilities beyond those which exist at WARD and are described in the Appendix. Some of these facilities will be expanded to meet the needs of this program.

7.5.6 Program Schedule

The schedule for this program is shown in Figure 7.5-1, with the milestones listed below.

Task 7.5.A - Sampling and Handling of Sodium

1. Initiate evaluation of sampling and handling techniques
2. Interim evaluation of sampling and handling results and begin the second phase of evaluations
3. Complete task, with selection of best techniques

Task 7.5.B - Determination of Oxygen Concentration and Species

1. Evaluate ongoing studies of analytical methods for oxygen analysis
2. Select reference and alternate method for oxygen analysis for further investigation
3. Complete task

Task 7.5.C - Determination of Carbon Concentration and Species

1. Evaluation of analytical methods for carbon analysis
2. Complete task and select reference method of analysis

Task 7.5.D - Determination of Hydrogen Concentration and Species

1. Evaluation of methods of analysis and species of hydrogen
2. Select a reference analysis method for sensitivity improvement
3. Complete task

Task 7.5.E - Determination of Nitrogen Concentration and Species

1. Initiate studies of analytical techniques
2. Select reference technique of analysis
3. Complete task

Task 7.5.F - Determination of Metallic and Halide Impurities

1. Evaluation of impurities and methods for analysis of each
2. Complete evaluation of analytical methods
3. Complete examination of hot and cold traps to insure adequacies of analytical techniques and species detection
4. Complete task

Task 7.5.G - Sampling Methods for Cold Traps and Hot Traps

1. Evaluation of sampling methods
2. Complete design of sampling equipment
3. Complete first series of sampling tests
4. Complete sampling of larger units

Task 7.5.H - Preparation of Standard Samples of Sodium

1. Initiate studies of standard preparation
2. Complete development of standard preparation system
3. Complete first series of test preparations
4. Complete standard preparation

Task 7.5.I - Nondestructive Chemical Analysis

1. Initiate instrument review for applicability
2. Make instrument adaptations for sodium analysis
3. Comparison of results of NDT instrumentation and selection for large scale plant

Figure 7.5-1. Sodium Sampling and Analysis Program Schedule

8. SAFETY AND ENGINEERED SAFEGUARDS

8.1 Radiation Shielding Development

8.1.1 Objective

To analyze and define the shielding requirements for the reactor and particular reactor components with respect to:

- a. Allowable biological dose rate limits
- b. Effects of nuclear heating
- c. Neutron-induced activation of reactor components or equipment
- d. Radiation damage effects

8.1.2 Program Summary

The program is divided into ten tasks. These include the preparation of basic nuclear and other physical data, the development of reliable calculational methods, and the shield design. In each case, theoretical as well as experimental investigations are involved. This applies to both neutron and gamma ray shielding problems. The tasks are listed below and described in the next section:

- Nuclear Cross Sections
- Determination of Physical Properties of Shield Materials
- Definition of Processing Codes for Nuclear Data
- Establishment of Methods of Performing Two-Dimensional Conventional Calculations
- Development of Transport Theory Codes
- Development of Three-Dimensional Calculational Techniques
- Determination of Release of Fission Products into the Coolant
- Determination of Secondary Sodium Activity
- Performance of Radiation Streaming Experiments
- Experimental Verification of Calculational Methods

8.1.3 Task Descriptions

Task 8.1.A - Determination of Nuclear Cross Sections

The objective is to obtain detailed data of differential microscopic cross sections of all fast reactor materials involved. Basic cross section measurements, evaluation, and compilation will be necessary. Special attempts will be given to evaluate higher moments of anisotropic scattering cross sections in the range of high and intermediate neutron energies. Some further experimental work will be necessary to improve and complete existing tables for neutron radiation capture cross sections in order to reach a reliable basis for the calculation of neutron-induced activation of reactor components and equipment.

Task 8.1.B - Determination of Physical Properties of Shield Materials

The objective is to obtain more data on certain physical properties of some shield materials, particularly those used for fast reactor shield construction, such as serpentine aggregate. The coefficients of thermal expansion and thermal conductivity are examples of those properties. In addition, the chemical interaction of sodium with certain hydrous constituents of the reactor shield should be tested experimentally. The development of radiation damage information is also part of this task.

Task 8.1.C - The Development of Definition of Processing Codes for Nuclear Data

The objective is to make basic nuclear data immediately applicable to advanced shielding calculations. For most applications, it is necessary to condense the experimental data of nuclear cross sections (Task 8.1.A) into a convenient energy group structure. Thus, complete cross section libraries will be established, directly matching the input of computer programs.

Task 8.1.D - Establishment of Methods of Performing Two-Dimensional Conventional Calculations

The objective is to establish preliminary calculations of the shielding requirements for the fast breeder reactor with well-known conventional

methods. Experimentally verified codes on the basis of multigroup diffusion theory in connection with a removal source calculation are available in the literature. To be useful as a design tool for fast reactor shielding calculations, it is necessary that the codes cover two-dimensional problems because of the more complex geometry arising from the more compact construction of fast breeder reactors in comparison with conventional thermal reactors. Existing codes of this type must be made operational on the CDC-6600, together with a complete library suitable for use with the code.

Task 8.1.E - Development of Transport Theory Codes

The objective is to establish a well-defined standard calculational tool for final design assessment of the overall neutron and gamma shield. The necessary codes will be based on multigroup transport theory in two-dimensional geometry. Special attention will be paid to the incorporation of typical transport effects, general anisotropic scattering, and a sufficiently good description of boundary conditions.

Task 8.1.F - Development of Three-Dimensional Calculational Techniques

The objective is to provide reliable calculational methods for the treatment of problems in complex geometries not covered by two-dimensional techniques. Those problems arise, for example, in connection with large ducts and voids within the shield for cooling requirements or with other gaps and irregularities in the shield.

Until satisfactory three-dimensional techniques, based on the general Boltzmann transport equation, are developed, one method of approach to the solution of those problems will be the comprehensive use of large Monte Carlo codes. The techniques must be general enough to incorporate neutron as well as gamma ray streaming calculation.

Task 8.1.G - Determination of Release of Fission Products into the Coolant

The objective is to determine, quantitatively, the release of fission products from the vented fuel to the coolant, and the rate of release and the subsequent transportation within the primary sodium loop.

Additional experimental studies seem to be the only way to obtain reliable answers to this problem. This task will be coordinated with Task 7.3 of this report.

Task 8.1.H - Determination of Secondary Sodium Activity

The objective is to reduce the sodium activity in the secondary loop to an extent that manned access requirements for the secondary circuit are accommodated without additional shielding for the secondary loop. In particular, extensive calculations in complex geometries (Task 8.1.F) combined with several refinements of the entire coolant system layout, will be necessary. Possible impurities of the secondary sodium coolant must be studied together with their neutron-induced activation. Special attention will be paid to those radioactive impurities having very large decay times.

Task 8.1.I - Performance of Radiation Streaming Experiments

The objective is to measure the radiation field of both neutron and gamma rays around ducts and voids in the shield where theoretical and empirical methods are not applicable or insufficient. At present, only few reliable methods exist for radiation streaming calculations in idealized ducts, slots, and annuli. Those calculations must be extended toward more realistic geometries supported by extensive measurements. In addition, existing empirical formulas for neutron and gamma ray streaming calculations will be improved and new ones stated.

Task 8.1.J - Experimental Verification of Computational Methods

The objective is to verify experimentally, under realistic conditions, the extended use of advanced calculational methods for the shield design. As long as no rigorous solution of the space- energy- and angular-dependent Boltzmann transport equation in three-dimensional geometry is known, each calculational method implies certain assumptions and approximations. The adequacy of those assumptions and the ranges of validity of approximations must be verified. In certain cases, pure theoretical verifications are insufficient because of the lack of either basic data or experience. For those cases, it is necessary to compare the

calculations directly with experiments. In general, compact high power fast breeder reactors are endowed with certain new features in their shielding requirements which need experimentation before applying theoretical predictions.

8.1.4 Relation to LMFBR Program Plan

The approach to the task of radiation shielding development within the LMFBR program plan is not very detailed and its explanation is spread over several volumes. It is recommended that radiation shielding development be incorporated in one section of the program plan. The general aspects and requirements, however, are in agreement with the foregoing plan.

The development of reliable calculational methods in conjunction with the production of efficient computer codes form the basis of reliable shield design. Therefore, Tasks 8.1.A, 8.1.C, 8.1.E and 8.1.F, together with Task 8.1.J, should be given priority.

8.1.5 Major Test Facility Requirements

The functional requirements and descriptive details of major test facilities, together with the ranges of operation and the required accuracy, are listed below. Most measurements could be obtained with the existing facilities slightly modified.

Nuclear Data Experiments (Task 8.1.A)

- a. The energy range for the required nuclear data is 0 to 18 MeV for both neutrons and gamma rays.
- b. The accuracy of differential scattering cross sections $\sum_g (E', \Omega' \rightarrow E, \Omega)$ (transfer cross sections) for neutrons should be of the order of ± 5 percent over the whole energy range. The differential scattering cross section for gamma rays needs no additional experimentation. It is analytically given by the well-known Klein-Nishina-Formula which is accurate enough within the whole energy range of interest.

- c. Group constants should range within ± 5 percent accuracy incorporating resolved and unresolved resonances.
- d. The desired accuracy of microscopic total cross sections is at least ± 1 percent.

Measurements of Physical Properties (Task 8.1.B)

This experimental facility can be integrated parts of larger facilities primarily designed to serve other purposes. Special attention however, must be paid to allow the following measurements:

- a. The coefficients of thermal expansion and thermal conductivity of various types of serpentine aggregate and serpentine concrete as well as the dependency of these properties on long-term (several years) intensive irradiation (up to about 10^{14} n/cm²s above 0.1 MeV).
- b. The dehydration of concrete and other hydrous materials, as a result of strong local heat production (up to 5 Watt/cm³) and the subsequent diffusion of hydrogen within the material, arising from the temperature profile.
- c. The density variation on various concretes undergoing long-term irradiation (several years) and heating (maximum local temperature rise 20°F).

Final Test Facility (Tasks 8.1.I, 8.1.J)

For the final testing and verification of computational methods, certain differential and integral flux measurements must be carried out at various shield configurations. This facility must be designed to allow the following measurements:

- a. Energy spectra of neutrons and gamma rays with a accuracy of ± 10 percent for attenuations up to 10^{-6} .
- b. Angular flux measurements with a resolution of $\Delta\Omega = 10^\circ$.
- c. Dose rate measurements with an accuracy of ± 10 to 20 percent for attenuations up to 10^{-6} .

The test facility must be able to produce a range of neutron spectra which simulate the various spectra encountered in fast reactor shielding. In particular, the facility should be capable of producing neutron spectra which have a high intensity in the region 50 keV to 500 keV.

8.1.6 Program Schedule

The major milestones for the program are shown in Figure 8.1-1, and are defined as follows:

Task 8.1.A - Determination of Nuclear Cross Sections

1. Measurements completed on basic nuclear cross sections.

Task 8.1.B - Determination of Physical Properties of Shield Materials

1. All required physical properties determined.

Task 8.1.C - Definition of Processing Codes for Nuclear Data

1. Complete nuclear data libraries available.

Task 8.1.D - Establishment of Methods of Performing Two-Dimensional Conventional Calculations

1. Completion of present studies.

Task 8.1.E - Development of Transport Theory Codes

1. Two-dimensional transport codes in operation.

Task 8.1.F - Development of Three-Dimensional Computational Techniques

1. Examination of complex geometries with Monte-Carlo techniques.
2. Three-dimensional techniques in operation.

Task 8.1.G and 8.1.H - Determination of Release of Fission Products into the Coolant and Determination of Secondary Sodium Activity

1. Experiments in deep penetration problems are initiated.
2. Completion of Experimental program.

Task 8.1.H - Performance of Radiation Streaming Experiments

1. Examination of complex geometries with Monte-Carlo techniques.
2. Completion of radiation streaming codes.
3. Completion of experiments and testing on duct-streaming problems.
4. Three-dimensional techniques in operation.
5. Completion of experimental program.

Task 8.1.J - Experimental Verification of Computational Methods

1. Experiments on deep penetration problems are initiated.
2. Completion of experiments and testing on duct-streaming problems.
3. Completion of experimental program.

TASK	TITLE	CALENDAR YEAR						
		1969	1970	1971	1972	1973	1974	1975
8.1.A	NUCLEAR CROSS SECTIONS (DATA 1)				▽ 1			
8.1.B	PHYSICAL PROPERTIES OF SHIELD MATERIALS (DATA 2)					▽ 1		
8.1.C	PROCESSING CODES FOR NUCLEAR DATA (DATA 3)						▽ 1	
8.1.D	TWO-DIMENSIONAL CONVENTIONAL CALCULATIONS (METHODS 1)	▽ 1						
8.1.E	TRANSPORT THEORY CODES (METHODS 2)	▽ 1						
8.1.F	THREE-DIMENSIONAL TECHNIQUES (METHODS 3)			▽ 1			▽ 2	
8.1.G	RELEASE OF FISSION PRODUCTS INTO THE COOLANT (DESIGN 1)		▽ 1					▽ 2
8.1.H	SECONDARY SODIUM ACTIVITY (DESIGN 2)		▽ 1					▽ 2

Figure 8.1-1. Radiation Shielding Development Program Schedule

TASK	TITLE	CALENDAR YEAR						
		1969	1970	1971	1972	1973	1974	1975
8.1.I	RADIATION STREAMING EXPER- IMENTS (DESIGN 3)			1	2	3	4	5
8.1.J	EXPERIMENTAL VERIFICATION OF CALCULATIONAL METHODS (DESIGN 4)		1			2		3

Figure 8.1-1. (cont'd) Radiation Shielding Development Program Schedule

8.2 Blast Shielding Development

8.2.1 Objective

To design suitable shielding around the reactor vessel to absorb the shock loading following a hypothetical core meltdown and compaction accident; to avoid shock damage to the primary containment structure which otherwise might allow release of radioactivity.

8.2.2 Program Summary

The program is designed to provide material data which is lacking at the present time, in particular, for newer shielding materials such as serpentine, to provide better mathematical modeling techniques, and to test proposed designs. Data for the shock energy which would result from a core meltdown and compaction accident would be forthcoming from the program defining the containment building. The three tasks of the program, listed below are described in the next section.

- Determination of Material Properties Under Deformation Investigation
- Experimental Testing of T.N.T. Explosive Effect
- End Effect Model Development

8.2.3 Task Descriptions

It is assumed that as a result of Tasks 8.3.E through 8.3.I the shock energy resulting from a core meltdown and compaction accident and the partitioning of this energy would be known.

Task 8.2.A - Determination of Material Properties Under Deformation Investigation

The objective is to obtain physical properties for blast shielding materials, including steels and serpentine aggregate, for use in blast energy absorption calculations.

Densities, thermal conductivities, and material strain and stress strengths for high deformation rates are required.

Task 8.2.B - Experimental Testing of T.N.T. Explosive Effects

The objective is to confirm blast calculations in proposed shield designs. The work may be done in conjunction with scaled down deformation damage studies for the containment building (Task 8.3.I). The tests should include newer materials such as serpentine. Failure to confirm deformation effects in blast shielding may result in over-design and in an over-estimation of the volume needed for blast shielding within the containment.

Task 8.2.C - End Effect Model Development

The objective is to be able to calculate blast effects in complex geometries, in particular, to be able to calculate end effects of the radial shielding. The task will include model development and model confirmation tests within blast facilities such as those employed in Task 8.2.B above. Failure to be able to calculate blast damage in complex geometries, such as nozzle blast shielding, may result either in blast shielding over-design or undiscovered weak spots. Failure to discover weak spots could provide preferential failure paths for the release of sodium and fission products in the event of a hypothetical core meltdown and compaction accident.

8.2.4 Relation to LMFBR Program Plan

Blast shielding development has not been included in the LMFBR Program Plan. The present proposal, in particular, in Task 8.2.A, is original.

8.2.5 Major Test Facility Requirements

The functional requirements, and descriptive titles, of the major test facility required are:

Facility for Investigation of Material Properties Under Deformation (Task 8.2.A)

Such a facility should be capable of measuring material properties while the materials are being subjected to rapid deformation. The facility, therefore, should be able to subject materials to high pressure values and high rates of change. Hydraulic or explosive means of applying

pressure should be considered. Temperature conditions should approximate those experienced under extreme accident conditions.

The facility should, therefore, have:

- a. The ability to operate at as high a temperature as possible, certainly over 500°F
- b. The ability to apply very rapid pressure changes
- c. The ability to test all blast shielding materials
- d. The ability to measure transient values of strain and stress strengths
- e. The ability to measure other relevant material properties, e.g., heat capacity, thermal conductivity, and densities while subjected to deformation.

It is assumed that explosive blast testing of shield designs in simple and complex geometries will be performed in the same facility as required for Task 8.3.I and that material properties under normal operating conditions may be obtained from the facility required for Task 8.1.B.

8.2.6 Program Schedule

The blast shielding development schedule is shown in Figure 8.2-1. A task-by-task description of the illustrated milestones is presented below:

Task 8.2.A - Determination of Material Properties Under Deformation Investigation

1. Initiate investigation of materials property information.
2. Complete design and construction of the test facility.
3. Complete data collection.

Task 8.2.B - Experimental Testing of T.N.T. Explosive Effects

1. Initiate Task
2. Complete design and construction of chemical explosive facility.
3. Complete tests on shielding materials.

Task 8.2.C - End Effect Model Development

1. Initiate complete complex geometry explosive studies.
2. Complete complex geometry explosive studies.
3. Complete complex geometry model development.

Figure 8.2-1. Blast Shielding Development Program Schedule

8.3 Containment Design Basis Development

8.3.1 Objective

To calculate peak transient pressure and temperature conditions which could arise from the design basis accidents for containment design. This will ensure that the containment will sustain the effects of the worst hypothetical reactor accident to an extent necessary to avoid undue risk to the health and safety of the public.

8.3.2 Program Summary

The design basis accidents for the containment are the following: 1) the credible design base accident (CDBA) taken to be the most severe credible radioactive sodium reaction, and 2) the hypothetical design base accident (HDBA) taken to be the most severe core meltdown and compaction accident leading to high reactivity insertion culminating in the generation of large internal pressures and the violent dispersion of core materials.

The program is divided into tasks with specific reference to these design basis accidents (and they are so noted). The first four tasks refer to the CDBA and the latter five to the HDBA. In each case, analytical and experimental investigations are involved. The nine tasks are listed below and described in the next section.

- Sodium Combustion Studies
- Determination of Sodium Spray Fire Data
- Aerosol Spread of Radioactivity Studies
- Engineering Safeguards Studies
- Fuel Collapse Studies
- Material Properties Studies
- Determination of Nuclear Properties
- Model and Analytical Method Studies
- Energy Partition Analysis

8.3.3 Task Descriptions

Task 8.3.A - Sodium Combustion Studies (CDBA)

The objective is to obtain experimental confirmation of published sodium data, and, at the same time, to develop an adequate mathematical model of a sodium fire. The data include flame temperatures, radiant emissivities, heats of combustion, threshold values of oxygen concentration and temperature, the dependency of burning rates upon oxygen concentration and temperature, temperature-dependent heat transfer coefficients, diffusion coefficients of the sodium vapor, and the amount of counterflow in containment volumes. The experiments should be planned so that an adequate mathematical model of a burning sodium pool may be developed, (i.e., the experimental geometry should be simple, initial conditions well defined, and disturbances single rather than multiple). This mathematical model will include a self-limitation of the fire by blanketing from the by-products, an energy release, a temperature distribution within the sodium pool, adequate variations of pool geometry, accurate representation of flame and air-combustion products mixture, and other phenomena representative of a burning sodium pool. The model will be developed with sufficient flexibility to ensure easy adaptation to a variety of situations regarding location, depth of pool, and interaction of the sodium with air, water, concrete, or other materials. Failure to accurately identify and analyze these situations may result in an over-estimation of the design of the containment building or vaults concerned with containing a sodium fire.

Task 8.3.B - Determination of Sodium Spray Fires Data (CDBA)

The objective is to obtain experimental data of burning rates for sodium fires under both a spray and an explosive ejection condition. The experimental data will be such that the dependency of the burning rate upon oxygen concentration, sodium and air temperatures, sodium particle size, humidity, etc., may be ascertained with sufficient accuracy to develop a mathematical model of the burning mechanism. The experiments will be conducted with the ability to vary particle size and to allow representation of a number of accidents ranging from a postulated spill

from a fuel handling machine, to a fine forced spray of a pin hole pipe break, to an ejection of a large amount of sodium spray due to a core meltdown. Failure to define an accurate mechanism by which a sodium spray fire propagates will ultimately result in over-design of the containment structure and other housings where spray fires are postulated to occur.

Task 8.3.C - Aerosol Spread of Radioactivity Studies (CDBA)

The objective is to obtain information regarding the spread of radioactivity by an aerosol mechanism following a sodium spray fire, and to prevent such an aerosol spread of radioactivity. Experimental investigations will be conducted to establish the relationships between the extent of a sodium spray, the amount, and the suspension time of the radioactive aerosol. Aerosol settling devices will be required, and a program for testing such devices will be instituted. (See Tasks 8.6.A and 8.6.B) Other mechanisms, such as gas-tight compartments, and the like, also will be investigated to prevent the spread of the radioactive mist.

Failure to define the problem accurately, and the means of controlling it, may result in over-design of clean-up systems.

Task 8.3.D - Engineering Safeguards Studies (CDBA)

The objective is to develop engineered safeguards to prohibit, inhibit, and terminate any sodium fire. These safeguards may include such items as:

- a. Restrictive vault geometries to prevent access of sodium to large volumes of air or to maintain possible sodium surface areas at a minimum
- b. The design of preferential flow mechanisms to insure that the restrictive vault geometries will be used when necessary (this may be possible with rupture disks located at strategic positions on the floor designed to direct sodium flow into sub-floor storage in the event of a spill)

- c. A sprinkling system for chemical fire protection designed to smother burning sodium pools.
- d. Large clean-up systems to replace contaminated atmosphere and to quickly inert an area.
- e. The settling of a sodium spray by an ultrasonic or electromagnetic device; and other mechanisms capable of achieving the result of either limiting, preventing, or extinguishing a sodium fire. Failure to establish working models of such mechanisms will result in over-design to ensure against the possibility of a fire.

Task 8.3.E - Fuel Collapse Studies (HDBA)

The objective is to obtain realistic information regarding the mechanisms for the collapse of fuel rods under extreme conditions and to be able to treat these mechanisms mathematically. Experimental investigations will be conducted to determine the fuel damage propagation and core collapse mechanisms under a variety of controlled conditions which will enable mathematical treatment of the phenomena. Such studies will take into account possible shock loadings when the fuel rods have lost structural integrity in addition to the effect of gravity. The studies also will take into account the presence or absence of coolant and the variation with time of the collapse mechanism. A realistic slumping model will enable calculation of realistic reactivity addition rates. Failure to accurately identify and analyze sodium ejection modes and fuel slumping mechanisms which might result in a secondary critical configuration leads to the use of grossly pessimistic initiating conditions for the core meltdown and compaction accident. Such pessimistic assumptions lead to an over-design of the containment. A short-term product of this study will be an accurate fuel failure criteria for use in control studies.

Task 8.3.F - Material Properties Studies (HDBA)

The objective is to establish physical, chemical, and mechanical properties of core materials under extreme accident conditions. Equations of state for both fuel materials and sodium will be developed. Interactions between fuel materials and sodium at very high temperatures must

be defined. These investigations will involve experimental facilities and subsequent calculational analysis, the final objective being to develop a complete library of material properties.

Task 8.3.G - Determination of Nuclear Properties (HDBA)

The objective is to establish nuclear properties of core materials under extreme accident conditions. The Doppler effect for fuel materials at very high temperatures must be defined in order to provide accurate feedback information during the the meltdown accident. Failure to obtain an accurate temperature dependence of the Doppler leads to pessimistic assumptions.

Task 8.3.H - Model and Analytical Method Studies (HDBA)

The objective is to establish a well-defined standard calculational tool for the treatment of a core meltdown and compaction accident. The work will include an assessment of autocatalytic effects and pressure waves present during the expansion mode. The task will require a realistic fuel slumping model and consequent nuclear representation from Task 8.3.E as the input.

Task 8.3.I - Energy Partition Analysis (HDBA)

The objective is to define the partition of energy resulting from the reassembly criticality. In particular, it will be necessary to give an assessment of work expended in vessel and shielding deformation, missile projection, and in the creation of transient conditions to define the containment. The work will include a determination of the energy required for the expansion of molten core materials within the vessel. Consideration will be given to possible experimental confirmation of the dispersal model. This task will require the results of Task 8.3.F before any analytic confirmation of experimental work is possible. Establishment of realistic energy partition patterns will avoid pessimistic assumptions and over-designed shielding and containment.

8.3.4 Relation to LMFBR Program Plan

The LMFBR Program Plan, Volume 10 "Safety," is in good agreement with the Westinghouse approach for treating a sodium fire, be it pool, spray, or explosive ejection. However, Westinghouse gives more stress to the need to extinguish or, at least, control a fire once it has begun.

The AEC approach to the task of obtaining information necessary for the correct treatment of the core meltdown and compaction accident is in good agreement with the foregoing program. Basic requirements of the AEC coincide with Westinghouse ideas.

The AEC accepts that the tasks have high priority and that the program, necessarily, is a very long term plan. It is noted, however, that short term objectives, such as fuel failure criteria, will be achieved.

The only criticism of the Program Plan is that there is not sufficient differentiation between the analytic and experimental approaches to the problem. Suitably designed experiments which are amenable to mathematical analysis can provide both an understanding of the mechanisms and a check on mathematical models.

It is recognized that experimental treatment of meltdown and dispersion mechanism is very costly, but the fulfillment of these task objectives is vital to the LMFBR Program.

8.3.5 Major Test Facility Requirements

The functional requirements together with the ranges of operation, where known, and descriptive titles of major test facilities are listed below.

Sodium Combustion Facility (Task 8.3.A)

To adequately determine the data required, the apparatus should consist of a mechanism to contain a pool of sodium (burn pot) with ample sensing devices to measure the flame temperature and the burning rate. This equipment will be flexible enough to vary the volume of the sodium, the geometry of the pool, the volume of the air in direct contact with the pool, the oxygen concentration of this air, and the air flow above the

pool. The facility will allow the confirmation of emissivity coefficients from flame-to-sodium, gas-to-wall, etc., and heat transfer coefficients as functions of temperature.

Pressure transducers will be located throughout the air compartment to sense variations in pressure during the experiment. The tests also will be conducted with varying pool temperatures to determine the threshold temperature needed for a sodium pool fire ignited by spontaneous combustion.

Sodium Spray Fire Facilities (Task 8.3.B)

These facilities will consist of a number of housings in which sodium spray fires may be initiated by different means. Variable sized enclosures will be used for these housings. The facilities will have the ability to record temperature and pressure histories during the sodium spray combustion.

Oxygen concentrations, sodium and atmosphere initial temperatures, and humidity will be varied. A number of experiments will be conducted in these facilities, ranging from dropping of sodium liquid from various heights into a sealed container to simulate sodium spilling from a refueling mechanism to an explosion in a pool of sodium to eject it into an area containing oxygen.

A nozzle arrangement also will be available for spray fire work, with a variable nozzle diameter and variable pressure, so as to spray sodium into the housing at a variety of velocities. By combining this nozzle arrangement with spillage and explosive mechanisms described above, sufficient flexibility to handle all types of sodium spray fires will be achieved.

Aerosol Facilities (Task 8.3.C)

The facilities for the study of aerosol behavior and settling are described in Section 8.6.5 of this report. The facilities will be comprehensive enough to fulfill requirements set forth under Task 8.3.C.

Safeguards Testing Techniques (Task 8.3.D)

To properly evaluate an engineered safeguard, a number of experiments

or tests must be conducted to ensure reliability when the safeguard is required. Items, such as vault geometries, and preferential flow mechanisms may be evaluated easily while normal burning rate experiments are being conducted without disturbance of the burning rate experiment. The Sodium Combustion Facility may be used. Tests on sprinkling systems, venting mechanisms, and collecting devices, however, must be conducted as single experiments in preferably large areas to simulate conditions where the situation is most likely to occur.

Fuel Slumping (Task 8.3.E)

This facility should attempt to reproduce fuel damage propagation within a fuel subassembly. It can take the form of an electrically heated facility since its function is to understand the failure and slumping mechanisms and provide a check on mathematical models. It should include the following:

- a. The ability to be cooled by liquid metal which may be drained out of the test assembly (flow up to 300 gpm).
- b. The ability to heat fuel to destruction by electrical heating (2000 watts/cm^3), and the ability to provide a non-linear heating distribution within the assembly both radially and axially.
- c. The ability to accept a number of fuel subassemblies up to 36 inches long and 12 inches outside diameter.
- d. The ability to provide a photographic history of the destruction and subsequent slumping.
- e. The ability to provide a temperature history of the event.

The fuel will cease to be heated after slumping, and, therefore, the time effect of continued nuclear heating will require analysis in a TREAT facility using a single fuel rod. It is recognized that FFTF may be able to treat subassembly destruction tests.

Material and Physical Properties Facilities (Task 8.3.F)

These facilities should be designed to operate in a temperature range and a pressure range as high as possible and yet be capable of adding

known quantities of heat to the sample. They will require specially designed procedures and test components, and they should be able to determine the following properties:

a. Density	}	of sodium and fuel materials
b. Enthalpy (heat addition)		
c. Specific heats		
d. Mechanical strengths	}	of fuel and structural materials
e. Thermal conductivity		
f. Viscosity		
g. Emissivity		

Nuclear Properties Facilities (Task 8.3.G)

These facilities will require a high temperature and high pressure loop in a fast flux assembly. They could involve a thermally insulated loop in a reactor, such as the FFTF, within which temperatures in excess of 2000°F might be produced. The facility also may take the form of a high temperature and high pressure test bed in an accelerator program. The required temperature range is to be to as high a limit as possible and, likewise, the maximum pressure is to be as high as possible.

Dispersion Facility (Task 8.3.I)

Possible experimental confirmation of the dispersion model will be considered within:

- a. A specially designed nuclear facility
- b. A chemical explosive facility in which applicability has been assessed

The nuclear facility components should be of simplified design to ease mathematical assessment of deformation energies following the explosion. It should be capable of measuring pressure waves of as high a magnitude as possible within and also outside the vessel. It should be designed to check more specifically mathematical mode techniques rather than to reproduce in detail any particular core.

The chemical explosive facility will be designed to provide deformation data for the analysis of the results of the nuclear explosion. The facility should cater for explosions up to scaled equivalents of 4000 tons of TNT.

8.3.6 Program Schedule

The containment design basis development schedule is shown in Figure 8.3-1. A task-by-task description of the illustrated milestones is presented below:

Task 8.3.A - Sodium Combustion Studies (CDBA)

1. Initiation of task
2. Completion of design and construction of a sodium pool fire facility
3. Completion of sodium combustion studies
4. Completion of a mathematical model to analyze a sodium fire

Task 8.3.B - Determination of Sodium Spray Fires Data (CDBA)

1. Initiation of task
2. Completion of a facility to study sodium spray and explosive ejection fires
3. Completion of spray fires studies
4. Completion of a mathematical model to analyze a sodium spray fire

Task 8.3.C - Aerosol Spread of Radioactivity Studies (CDBA)

1. Initiation of task
2. Completion of the aerosol spread of the radioactivity studies
3. Completion of the design and construction of devices to prevent spread of radioactivity due to an aerosol

Task 8.3.D - Engineering Safeguards Studies (CDBA)

1. Initiation of task
2. Completion of the design, construction, and tests of safeguards against sodium fires

Task 8.3.E - Fuel Collapse Studies (HDBA)

1. Initiation of task
2. Completion of the design and construction of the fuel slumping facility
3. Completion of the fuel collapse studies

Task 8.3.F - Material Properties Studies (HDBA)

1. Initiation of task
2. Completion of the design and construction of new test facilities
3. Completion of the material property investigation

Task 8.3.G - Determination of Nuclear Properties (HDBA)

1. Initiation of task
2. Completion of the nuclear facility for high temperature nuclear measurements
3. Completion of the nuclear measurements

Task 8.3.H - Model and Analytical Method Studies (HDBA)

1. Initiate task
2. Completion of the analysis model for fuel slumping Task 8.3.E
3. Completion of the HDBA model

Task 8.3.I - Energy Partition Analysis (HDBA)

1. Initiate task
2. Completion of the construction of a chemical explosion facility
3. Completion of the design and construction of a nuclear dispersion facility
4. Completion of scaled chemical explosion studies
5. Completion of nuclear dispersion studies

Figure 8.3-1. Containment Design Basis Development Program Schedule

8.4 Protection Against Loss of Coolant

8.4.1 Objective

To prevent loss of coolant from the reactor core by:

- a. The prevention of a pipe rupture in critical areas of the loop
- b. The provision of engineered safety features to prevent loss of coolant from the core in the event of a pipe rupture

8.4.2 Program Summary

The hypothetical design basis accident for the reactor cooling system is the circumferential rupture of the pipe, assuming unobstructed discharge from both ends. If left unchecked, the design basis assumes gross loss of coolant from the core.

The program will analyze both the necessity and effectiveness of the methods to prevent pipe rupture and, if the rupture should occur, to prevent subsequent loss of coolant.

The program is divided into six tasks. Taken together, they will provide the experimental data and mathematical analysis necessary to ascertain the design parameters of the engineered safeguards. In addition, they will provide means for testing siphon breakers, check valves, and the secondary containment of pipe work.

8.4.3 Task Descriptions

Task 8.4.A - Determination of Pipe Failure Modes

The objective is to obtain sufficient data on pipe failure modes so that areas in the primary loop of particular susceptibility may be defined and the probability of pipe rupture may be brought into proper perspective.

This task will serve to define the magnitude of the pipe rupture problem and help to establish the parameters necessary for the detailed analysis of the results of pipe rupture in the particular case of low pressure, high temperature liquid metal pipe loops as against high pressure water loops.

For clarity, the task is further divided as follows:

- a. An extension of present analysis in the area of pipe failure modes to determine areas of interest and those areas where more investigation is needed (i.e. thin-walled pipe in a high temperature, low pressure application).
- b. Comparison of a pipe rupture before the primary coolant pump with a rupture beyond the primary coolant pump in order to evaluate the effects of rupture location on the accident consequences.
- c. Development of a mathematical model for the modes of rupture to describe the results of the experimental investigation and the subsequent incorporation into the primary loop rupture model (Task 8.4.C).

Task 8.4.B - Double Containment Piping Studies

The objective is to test the necessity, applicability, and failure modes of double containment piping. Based on the results of this study, the necessity for limited double containment pipework in critical areas may be assessed.

Task 8.4.C - Formulation of Primary Loop Rupture Models

The objective is to formulate an accurate mathematical model of the reactor coolant boundary with provision for introducing a rupture of any size. It will take into account such things as the thermal-hydraulic and nuclear parameters associated with the plant and make provision for transient analysis of their variations during a pipe rupture. It must accurately describe the rupturing procedure and the effect of the rupture on the vessels, pumps, and intact loops. It must have, moreover, provision for the inclusion of the engineered safety features designed to mitigate the effects of the rupture.

This program will provide the parameters required to define the necessity, magnitude, response, and effects of the engineered safety features. This will preclude pessimistic parameters being applied in the design of the engineered safety features and the possible inclusion of unnecessary safeguards.

A program of experimental confirmation of the primary loop rupture models will be initiated using scaled, simple flow loops. This will provide a basis of correlation among the models and the necessary experimental data. The loop model will require rupture analysis resulting from Tasks 8.4.A, 8.4.B, 8.4.D, and 8.4.F.

Task 8.4.D - Sodium Hammer Investigations

The objective is to establish, experimentally, sodium hammer effects on the primary system.

Analysis will be conducted to determine the effect of check valve closure, double-ended pipe rupture, and pump seizure. Particular attention will be given to the effect of a check valve closure with and without attendant pipe rupture.

The investigation will include determination of the pressures and pressure differentials, resulting flow transients, void formation, and collapse in the loop which experiences the rupture or valve closure and in the remaining intact loops. The stress produced in the piping, particularly Type 304 SS, will be investigated.

This experimental analysis is necessary to complete the primary loop rupture model (Task 8.4.C) and to describe accurately the total effects of a pipe rupture on the primary system. The facility must be designed to be amenable to a mathematical analysis (i.e. the experimental geometry should be simple, initial conditions well-defined, and the disturbance single rather than multiple).

Task 8.4.E - Thermal-Hydraulic Model Studies

The objective is to develop a model and computer code that will describe accurately the thermal-hydraulic behavior of the primary loop during all normal operating conditions and abnormal but non-rupture conditions. The code must describe accurately the transients resulting from a change in load, including startup and trip operations, and also from changes in pump flows.

The model and code must be capable of describing operations with one or more loops isolated.

Particular attention will be paid to the effects of valving and flow values in each loop with provision in the code for the opening, closing, and adjustment of valves.

All components of the primary loop including pumps, heat exchangers, the reactor vessel, bends, elbows, makeup, and bleed flows must be mathematically described as accurately as possible. Also, it must accurately represent pressure equalization among loops and pressure transients.

Task 8.4.F - Siphon Breaker Studies

This task has two objectives:

- a. To confirm experimentally the operation and efficiency of the proposed siphon breaker systems under high pressure differentials.
- b. To obtain sufficient data on the erosion of valve seatings in sodium to check the feasibility of siphon breakers allowing percentage bypass flow in normal operation.

The siphon breaker systems can be divided into two categories: 1) those with valving, and 2) those which use some other method (i.e. sodium bypass channels) to break the siphon.

As to the former, work will include assessment of the time delay in valving actuation as a function of pressure differential and the time delay between the actuation of the valve and the breaking of the siphon. Particular attention will be paid to the amount of sodium lost before the siphon is broken.

In the latter, the time delay and amount of sodium lost are also important considerations and must be evaluated. In addition, the amount of bypass required must be evaluated along with the reliability of the system.

In order to compare valve seating erosion versus bypass flow, the amount of bypass that can be tolerated under normal conditions will be determined as a function of the valve lifetime. This will affect the time required to break the siphon.

The parameters determined in this task then will be applied to Task 8.4.C to further refine the mathematical model.

8.4.4 Relation to LMFBR Program Plan

The LMFBR Program Plan approach to the task of obtaining the necessary information for the design of engineered safety features in Volume 10 is in general agreement with Westinghouse ideas.

The Program Plan is not specific, but it accepts the need for adequately designed engineered safety features to prevent loss of coolant.

The program presented here attempts to define tasks explicitly and to provide the necessary safeguard design criteria. The supporting experiments will provide adequate confirmation of the analytical results and enable refinement of the mathematical models.

8.4.5 Major Test Facility Requirements

The functional requirements and descriptive titles of major test facilities, together with operating ranges where known, are listed below.

Pipe Rupture Mode Facility (Task 8.4.A)

Existing pipe rupture facilities may be utilized with modifications to handle stainless steel piping, up to 24 inch OD and 1/2 inch wall, containing low pressure and high temperature sodium.

Double Containment Piping Facility (Task 8.4.B)

This facility should be capable of analyzing the failure modes of a section of double containment piping where size is similar to that described in the above Facility. Conceivably, it could be a part of that Facility with the proper modifications, but it probably would be appropriate to design a short section, special purpose rupture facility.

Primary Loop Rupture Model Facilities (Task 8.4.C)

These facilities will consist of the following:

- a. A scaled, simple flow loop with typical components, i.e. pump,

vessel, IHX, cover gas, and associated piping, represented in terms of their gross parameters, i.e. flow characteristics, pressure drops, etc. Provision will be included for introducing a pipe rupture of any size up to a double-ended rupture with unobstructed discharge from both ends. The flows and piping must be of such a size that the flow characteristics of sodium flows up to 40 ft/sec in 24 inch OD piping are well simulated.

The rupture may be simulated by the application of an external force to a section of pipe which has been joined and sealed, in such a manner that the seal may easily be broken to produce the necessary pipe rupture.

- b. A scaled, multiple loop model which incorporates, as accurately as possible, the components of the full-sized plant with the provision to initiate a pipe rupture on one loop. Provision must be made to measure the effect on the unruptured loop. The loop should operate between 700°F and 1000°F and between 140 psia and 1 in. Hg.

In both facilities, instrumentation must be provided to measure and record flow, pressure, and temperature transients as well as the occurrence of sodium vapor. Photographic techniques may be used.

Sodium Hammer Facility (Task 8.4.D)

The facility should attempt to reproduce the effects of sodium hammer on primary system piping and components.

It should be a scaled, two-loop facility and, conceivably, experimentation of sodium hammer could be included in the Primary Loop Rupture facility with suitable modifications and application of the proper instrumentation.

Siphon Breaker Facility (Task 8.4.F)

The facility should consist of a scaled model of the reactor vessel, with its associated inlet and outlet piping, and make provision to include the siphon breaker valve system.

Preferably, the siphon should be set-up by initiating a double-ended pipe rupture.

Instrumentation should be provided to record the flow transients experienced and the response of the siphon breakers. With the proper modifications, use could be made of the Primary Loop Rupture Facility.

8.4.6 Program Schedule

The protection against loss of coolant program schedule is shown in Figure 8.4-1. A task-by-task description of the illustrated milestones is presented below.

Task 8.4.A - Determination of Pipe Failure Methods

1. Initiation of task.
2. Completion of modifications on existing pipe failure test facilities.
3. Completion of testing of pipe failure modes and development of a mathematical model.

Task 8.4.B - Double Containment Piping Studies

1. Initiation of task.
2. Completion of construction (or modification) of test facility.
3. Completion of testing of double containment pipe failure modes.

Task 8.4.C - Formulation of Primary Loop Rupture Models

1. Initiation of task.
2. Completion of construction of a single loop scaled model.
3. Completion of a basic primary loop rupture model for single and multiple loops.
4. Completion of construction of multiple loop scaled model.
5. Completion of testing of pipe ruptures of varying severity on single loop and subsequent model modifications.

6. Completion of testing of pipe ruptures of varying severity on multiple loop and subsequent model modifications.

Task 8.4.D - Sodium Hammer Investigations

1. Initiation of task.
2. Completion of construction of sodium hammer facility or modification of existing facility.
3. Completion of sodium hammer investigation with incorporation of results into pipe rupture and thermal-hydraulic models.

Task 8.4.E - Thermal-Hydraulic Model Studies

1. Initiation of task.
2. Completion of the thermal-hydraulic model.

Task 8.4.F - Siphon Breaker Studies

1. Initiation of task.
2. Completion of construction of siphon breaker test facility or modification of existing facility.
3. Completion of analysis of siphon breaker performance and inclusion of results into pipe rupture model.

Figure 8.4-1. Protection Against Loss of Coolant Program Schedule

8.5 Protection Against Secondary Criticality

8.5.1 Objectives

To be able to design large LMFBR-reactor cores and vessel structures exhibiting inherent design features which prevent the forming of a secondary critical configuration after core meltdown.

To find means ensuring the reliable dispersal, storage, and disposal of the molten fuel debris after core meltdown, incorporating cooling capacity.

8.5.2 Program Summary

There is a close relationship between the methods which aim at the prevention of secondary criticality and the postulations and design features which determine the limits of the energy release after the hypothetical design base accident (HDBA). The present program, therefore, to some extent overlaps with the one dealing with the containment building accident conditions (Section 8.3 of this report). The main difference is due to the fact that the present program deals not only with the meltdown accident itself, but also with the secure and reliable disposal of the highly radioactive molten fuel mass which remains after the meltdown.

The tasks under this program include those dealing with the meltdown mechanism and the prevention of secondary criticality, and those dealing with the dispersal, storage, and disposal of the molten fuel mass.

8.5.3 Task Description

Task 8.5.A - Fuel Collapse Studies

This task is similar to Task 8.3.E. The objective is to obtain realistic information regarding the mechanisms for the collapse of fuel rods under extreme conditions and to be able to treat these mechanisms mathematically. However, apart from the accurate prediction of fuel slumping and reactivity insertion rates in existing designs of core structures and fuel assemblies, the task under this program should be directed towards the goal of designing core structures and fuel assemblies which

inherently prevent the forming of a secondary critical configuration. Fuel rod configurations will be designed to meet this criterion. The effects of coolant flow, shock loadings, sodium ejection, thermal cycling, and fatigue mechanisms will be evaluated. The possibility of incorporating structural design features which control the motion of the fuel during meltdown in such a way that the forming of a supercritical configuration is prohibited, will be investigated.

Task 8.5.B - Nuclear and Material Properties Studies

Nuclear, physical, chemical, and material properties of core materials and of possible structural materials for core dispersal and terminal cooling systems will be established. There is, again, a considerable similarity with Tasks 8.3.D and 8.3.E. The nuclear and material properties libraries established under those tasks shall be extended to include those materials which are of interest for the design of dispersal systems and terminal cooling systems.

Task 8.5.C - Fuel Motion Studies

Methods will be developed which will enable the prediction and control of the movement of the fuel through a dispersal system. These methods will make use of mathematical models to calculate the temperature distribution and reactivity of moving fuel configurations and the energy balance between these configurations and their environment. The final distribution of the molten fuel as a result of passage past the dispersal system will be evaluated.

Task 8.5.D - Cooling Methods Development

Adequate cooling methods will be developed which will enable the reliable storage and disposal of the molten fuel debris after meltdown and passage through the dispersal system. Studies under this task will include an assessment of the heat generation as a function of time in the molten fuel mass; an assessment of the optimal cooling during and after the movement of the fuel; an assessment of heat sink configurations and materials, and the heat transfer mechanisms which should be used to result in optional cooling of the fuel.

8.5.4 Relation to LMFBR Program Plan

The Program Plan approach to the problem of designing reliable core dispersal and terminal cooling systems is in good agreement with the present program. The Program Plan places high priority on the task of obtaining an understanding of fuel failure mechanisms and of the movement and the temperature history of the fuel mass after meltdown.

8.5.5 Major Test Facility Requirements

The functional requirements and descriptive titles of major test facilities are listed below, together with the ranges of operation where known.

Due to the close relationship between the present program and the one described under Section 8.3, the test facility requirements of these programs turn out to be essentially similar. A combination of the requirements under these two programs will, therefore, certainly present an economic advantage.

Fuel Slumping Facility (Task 8.5.A)

This facility will be the same as the one defined under Section 8.3.5. Functional requirements may be found in that section. The actual fuel subassemblies which are to be tested may be adapted to the specific requirements of Task 8.5.A under the present program.

Material and Physical Properties Facility (Task 8.5.B)

Functional requirements for these facilities are to be found in Section 8.3.5.

Nuclear Properties Facility (Task 8.5.C)

Functional requirements for these facilities are to be found in Section 8.3.5.

Fuel Motion Simulation Facility (Tasks 8.5.D and 8.5.E)

This facility will present the possibility of simulating the movement of the molten fuel through the dispersal system and the cooling of the fuel mass in the last ditch cooling system. It will consist of a simplified scaled down version of the reactor vessel, core internals, reactor coolant,

dispersal devices, refractory materials, cooling systems, and heat sink configurations, and it will present the possibility of studying the effects of passing a simulated high temperature fuel mass through this dispersal and cooling device. The facility will operate up to a temperature of 2500°F and will be able to handle simulated fuel material.

.5.6 Program Schedule

he protection against secondary criticality schedule is shown in Figure 8.5-1. task-by-task description of the illustrated milestones is presented below.

Task 8.5.A - Fuel Collapse Studies

1. Initiation of task.
2. Completion of the design and construction of the fuel slumping facility.
3. Completion of fuel collapse studies.

Task 8.5.B - Nuclear and Material Properties Studies

1. Initiation of task.
2. Completion of the design and construction of new test facilities.
3. Completion of nuclear facility for high temperature nuclear measurements.
4. Completion of material property investigations.
5. Completion of the nuclear measurements.

Task 8.5.C - Fuel Motion Studies

1. Initiation of task.
2. Design and construction of fuel motion simulation facility completed.
3. Development of methods to control fuel motion during and after meltdown completed.

Task 8.5.D - Cooling Methods Development

1. Initiation of task.
2. Design of heat sink of cooling capacity, to be incorporated in fuel motion simulation facility.
3. Construction of fuel motion simulation facility completed.
4. Development of optimal cooling strategy during and after meltdown completed.

Figure 8.5-1. Protection Against Secondary Criticality Program Schedule

8.6 Radiological Protection

8.6.1 Objective

To develop an engineered safeguard to provide a reliable and rapid method of removing radioactive aerosols which might arise following a design basis accident, in order to ensure that such aerosols do not hamper speedy remedial clean-up measures or contribute any additional risk to the health and safety of the public.

8.6.2 Program Summary

The design basis accidents are the sodium fire and the hypothetical meltdown and compaction accident. Both could result in radioactive aerosols which could be composed of sodium-oxygen products, fuel materials, or sodium.

The Task 8.6.B program is designed to provide an understanding of inherent settling mechanisms and to develop some engineered safeguards to speed the settling and clean-up process.

8.6.3 Task Descriptions

Task 8.6.A - Aerosol Settling Mechanism Investigation

The objective is to provide information on the basic processes in aerosol particle conglomeration and settling. Experimental investigations will be conducted to provide settling data for very small particle sizes (1 to 2 microns) which may give useful mathematical correlations. The experiments should take a range of temperature conditions, atmosphere movement, and pressure conditions into account. Such studies will provide a basic understanding of aerosol behavior which is necessary for the design of aerosol settling devices. The design might, otherwise, be entirely dependent upon experimental tests, and extrapolations to different conditions would be difficult.

Task 8.6.B - Aerosol Settling Device Development

The objective is to test various aerosol settling devices and to assess their relative and absolute efficiencies. This task has a number of

inherent subtasks which are the design of the different devices to be tested. They will include:

- a. High volume filtering systems
- b. Electrostatic deposition systems
- c. Sonic deposition systems
- d. Mechanical precipitators
- e. Thermal precipitators

The design and testing of such devices will take into account the likely materials involved as well as the pressure and temperature ranges in which the device may be required to operate following a design basis accident. They should also take into account the subsequent disposal of the radioactive waste fallout and plateout.

8.6.4 Relation to LMFBR Program Plan

The LMFBR Program Plan approach to the task of reliably and speedily removing radioactive aerosols is in good general agreement with the above program. It is apparent, however, that the art is very new and, therefore, the Program Plan requirements are stated only in very general terms. We have attempted to be more precise in task definition.

8.6.5 Major Test Facility Requirements

The required test facilities will include testing facilities for specific devices including those enumerated in Task 8.6.B. These facilities are not defined, and attention is paid only to facilities required for the program.

Aerosol Behavior Facility (Task 8.6.A)

A facility which allows the measurement of aerosol movement and settling for a variety of particle sizes and environmental conditions.

The facility should have the following capabilities:

- a. The ability to produce aerosols with particles sized in the range of 1 to 3 microns in diameter.

- b. The ability to reproduce temperature and pressure conditions which are possible following design basis accidents. Temperature range 0 to 500°F, pressure range up to 40 psia.
- c. The ability to include a number of gaseous environments including air and the inert gases and to induce currents of movement within these atmospheres in a reproducible manner.
- d. The ability to provide a spatial history for the movement of the aerosol by photographic or by electronic means.
- e. The ability to provide a size history for the conglomeration of the aerosol particles by photographic or other means.

Aerosol Settling Device Comparison Facility (Task 8.6.B)

A facility within which the efficiency of various settling devices may be compared; therefore, it should attempt to reproduce post-accident conditions. The facility should include capabilities (a) to (c), as in the Aerosol Behavior Facility, and in addition, it should have the ability to assess the efficiency of an aerosol settling device on the basis of rate of settling and total mass of aerosol handled.

8.6.6 Program Schedule

The radiological protection program schedule is shown in Figure 8.6-1. A task-by-task description of the illustrated milestones is presented below.

Task 8.6.A - Aerosol Settling Mechanism Investigation

- 1. Initiation of task.
- 2. Completion of the designed construction of the aerosol behavior test facility.
- 3. Completion of studies of aerosol behavior and provision of mathematical correlated understanding of the settling mechanism.

Task 8.6.B - Aerosol Settling Device Development

- 1. Initiation of task.

2. Completion of the design of aerosol settling devices which are to be compared.
3. Completion of the design and construction of the comparison test facility.
4. Completion of studies and choice of aerosol settling device.

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Figure 8.6-1. Radiological Protection Program Schedule

9. CORRESPONDENCE OF TASKS TO LMFBR PROGRAM PLAN

The research and development tasks described in this report are referenced to a corresponding task(s) or task area(s) in the LMFBR Program Plan.

These references are summarized in the following table.

TABLE 9-1

CORRESPONDENCE OF R&D TASKS TO THE LMFBR PROGRAM PLAN

	<u>Page in this Report^a</u>	<u>LMFBR Program Plan, Volume</u>	<u>LMFBR Program Plan, Task Area or Tasks</u>
2. Plant Design Studies	3-4		
2.1 Conceptual Design Studies	3-5	2	2-1, 2-2
2.2 Demonstration Plant Program	3-20	2	2-3
3. Fuel and Materials	3-25		
3.1 Fuel Rod Development	3-25	7	7-5 thru 7-10
3.2 Core Structural Materials Development	3-44	5 6 7	5-1 6-2, 6-3 7-11, 7-12
3.3 Steam Generator Materials Development	3-53	3 5	3-6 5-1
3.4 Intermediate Heat Exchanger Materials Development	3-62	3 5	3-12 5-1, 5-3
3.5 Large Sodium Pump Materials Development	3-67	3 5 7	3-11 5-1
3.6 Check Valve Materials Development	3-73	3 5	3-8.5 5-1
3.7 Piping "Catastrophic" Corrosion Studies	3-79	5	5-1

TABLE 9-1 (Continued)

	Page in this Report ^a	LMFBR Program Plan, Volume	LMFBR Program Plan, Task Area or Tasks
4. Reactor Development	3-82		
4.1 Studies of Temperature Coefficient of (U, Pu)C Fuel	3-82	6 9	6-1 9-1, 9-4
4.2 Development of Na Fuel Bond	3-89	6 10	6-2 10-2.8, 10-2.9
4.3 Determination of Hydraulic Effects on Fuel Cladding	3-96	6 7 10	6-2 7-2, 7-7.2.1, 7-9 10-2.4
4.4 Studies of the Behavior of the Na Coolant	3-101	6 10	6-2 10-2.3, 10-5
4.5 Development of Fuel Assembly	3-111	6	6-1.2, 6-2.3
4.6 Development of a Radial-Blanket Assembly	3-118	6	6-1.2, 6-2, 6-3.1
4.7 Development of Control Rods and Drives	3-125	6	6-3.2
4.8 Studies of Modular Reactor Geometry	3-132	6 9 10	6-1 9-3 10-2.2
4.9 Development of Reactor Vessel Internals	3-138	6	6-1.3, 6-3.3
4.10 Development of Computer Assisted Fuel Management System	3-144	6 9	6-1.2.4 9-3.4.3
4.11 Development of a Non-Modular Core	3-151	6 10	6-1 10-2.2

TABLE 9-1 (Continued)

	Page in this Report ^a	LMFBR Program Plan, Volume	LMFBR Program Plan, Task Area or Tasks
5. Plant Systems and Components	3-156		
5.1 Reactor Vessel Development	3-156	3	3-9
5.2 Main Transport Piping Development	3-163	3	3-8
5.3 Check Valve Development	3-170	3	3-8.5
5.4 Large Sodium Pump Development	3-178	3	3-11
5.5 Sodium-to-Sodium Heat Exchanger Development	3-188	3	3-12
5.6 Steam Generator Development	3-198	3	3-6
5.7 Fuel Handling and Maintenance System Development	3-213	3	3-7
6. Instrumentation and Control	3-226		
6.1 Instrumentation Development	3-226	4	4-2, 4-3, 4-5, 4-6, 4-7
6.2 Control Systems Development	3-244	4	4-3, 4-4
7. Sodium Technology	3-253		
7.1 Liquid Metal Purification System	3-253	5	5-2
7.2 Cover and Hot Cell Gas Purification System	3-259	5	5-3.3, 5-3.6, 5-7

TABLE 9-1 (Continued)

	<u>Page in this Report^a</u>	<u>LMFBR Program Plan, Volume</u>	<u>LMFBR Program Plan, Task Area or Tasks</u>
7.3 Development of Non-Gaseous Fission Product and Active Corrosion Product Management Schemes	3-264	5	5-3
7.4 Sodium Quality Control	3-271	5	5-4
7.5 Sodium Sampling and Analysis	3-275	5	5-5
8. Safety and Engineered Safeguards	3-283		
8.1 Radiation Shielding Development	3-283	9 10	-
8.2 Blast Shielding Development	3-293	10	-
8.3 Containment Design Basis Development	3-298	10	-
8.4 Protection Against Loss of Coolant	3-310	10	-
8.5 Protection Against Secondary Criticality	3-319	10	-
8.6 Radiological Protection	3-325	10	-

a. These pages refer to Section III of this report.

SECTION IV
PROGRAM COSTS

1. INTRODUCTION

This report covers three types of costs to implement the program. These are:

- a. Costs of manpower, materials, and services to perform each of the program tasks, other than the Demonstration Plant Program
- b. Costs of facilities
- c. Costs of the Demonstration Plant Program

These cost estimates, and the ground rules, limitations, and exclusions of the estimates are presented in Section IV (2, 3, and 4 below).

2. DEVELOPMENT PROGRAMS

This section provides manpower, materials, and services costs for the programs identified in Section III, other than Task 2.2, Demonstration Plant Program. Estimates are provided to the second and, in some cases, third tier of task identification, resulting in a task estimating structure of fifty-nine line items.

The assumptions and ground rules in costing are as follows:

- a. Manpower estimates are calculated to even man-year, by categories of engineer-scientist and technician-draftsman
- b. Materials and services include experimental hardware and materials, computer time, and supplies unique to the development program (as opposed to normal operating and maintenance costs which are included in overhead).

- c. Rates are approximately equal to that represented by average 1969 rates for AEC contract work, including overhead and other adders. Escalation is not added to future year costs
- d. Cost of operation of AEC facilities, such as SPTF and SCTI, are not included
- e. Irradiation and loop rental costs in AEC test reactors are not included
- f. Lease or use charges on special nuclear materials used in development activities are not included

Table 2-1 summarizes manpower and cost estimates. Total engineer-scientist and technician-draftsman effort in man-years is tabulated for each task, together with materials and services estimates, and total estimates.

TABLE 2-1

SUMMARY OF MANPOWER AND COST ESTIMATES
FOR R&D TASKS

<u>Task No.</u>	<u>Title</u>	<u>E&S MY</u>	<u>T&D MY</u>	<u>M&S \$1000</u>	<u>Total Program \$1000</u>
2.	PLANT DESIGN STUDIES				
2.1	Conceptual Design Studies				
2.1.1	1000 MWe LMFBR Follow-on Study				--
2.1.2	Other Design Studies				
	Design Devel. Studies	70	10	130	3,190
	General Plant Studies	35	6	70	1,620
2.1.3	Demonstration Plant Conceptual Studies				
2.2	Demonstration Plant Program	-Cost Estimates in Sec. IV, 4-			
3.	FUEL AND MATERIALS				
3.1	Fuel Rod Development				
3.1.A	Evaluation of Fuel Fabrication Processes	25	30	600	2,250
3.1.B	Fuel Properties Determination	20	24	300	1,580
3.1.C	Cladding Properties (Unirradiated) Determination	12	8	180	840
3.1.D	Fuel-Cladding Compatibility Studies	10	10	160	780
3.1.E	Development of Fuel Rod Fabrication Procedures	18	20	460	1,620
3.1.F	Fast-Flux Rod Irradiation	30	40	800	2,860
3.1.G	Thermal-Flux Rod Irradiation	18	24	400	1,640
3.1.H	Transient Rod Testing	10	14	240	940
3.1.I	Cladding Properties (Irradiated) Determination	14	10	160	950
3.2	Core Structural Materials Development	6	4	180	410
3.3	Steam Generator Materials Development	17	12	200	1,150
3.4	Intermediate Heat Exchanger Materials Development	3	2	60	225
3.5	Large Sodium Pump Materials Development	2	1	20	125
3.6	Check Valve Materials Development	2	1	20	125
3.7	Piping "Catastrophic" Corrosion Studies	1	1	15	80
4.	REACTOR DEVELOPMENT				
4.1	Studies of Temperature Coefficient of (U,Pu)C Fuel	12	4	120	710

TABLE 2-1 (Continued)

<u>Task No.</u>	<u>Title</u>	<u>F&S MY</u>	<u>T&D MY</u>	<u>M&S \$1000</u>	<u>Total Program \$1000</u>
4.2	Development of Na Fuel Bond	9	4	120	580
4.3	Determination of Hydraulic Effects on Fuel Cladding	30	12	300	1,600
4.4	Studies of the Behavior of the Na Coolant	22	14	400	1,600
4.5	Development of Fuel Assembly	48	26	540	3,360
4.6	Development of a Radial-Blanket Assembly	24	16	180	1,510
4.7	Development of Control Rods and Drives	18	12	240	1,240
4.8	Studies of Modular Reactor Geometry	5	4	60	350
4.9	Development of Reactor Vessel Internals	42	22	260	2,460
4.10	Development of Computer Assisted Fuel Management System	6	2	120	410
4.11	Development of a Non-Modular Core	15	6	250	1,000
5.	PLANT SYSTEMS AND COMPONENTS				
5.1	Reactor Vessel Development	32	15	320	1,960
5.2	Main Transport Piping Development	14	5	150	840
5.3	Check Valve Development	26	12	400	1,740
5.4	Large Sodium Pump Development	42	20	1,200	3,360
5.5	Sodium-to-Sodium Heat Exchanger Development	36	18	500	2,380
5.6	Steam Generator Development	90	45	2,400	7,100
5.7	Fuel Handling and Maintenance System Development	40	20	800	2,930
6.	INSTRUMENTATION AND CONTROL				
6.1	Instrumentation Development				
6.1.A	Nuclear Instrumentation Development	12	5	300	910
6.1.B	Temperature Instrumentation Development	18	3	200	600
6.1.C	Coolant Flow Instrumentation Development	14	4	220	890
6.1.D	Pressure Instrumentation Development	5	2	140	390
6.1.E	Development of Strain Measurement Equipment	5	2	110	460

TABLE 2-1 (Continued)

<u>Task No.</u>	<u>Title</u>	<u>E&S MY</u>	<u>T&D MY</u>	<u>M&S \$1000</u>	<u>Total Program \$1000</u>
6.1.F	Development of Level Measurement Equipment	3	1	60	200
6.1.G	Fission Product Detection Development	8	2	160	540
6.1.H	Leakage Detection Development	6	2	140	430
6.1.I	Development of Signal Transmission Components	4	1	50	240
6.2	Control Systems Development				
6.2.A	Reactor Control System Development	20	10	150	1,190
6.2.B	Plant Control Systems Development	24	16	200	1,590
6.2.C	Development of Digital Control and Data Logging System	6	4	80	410
6.2.D	Reactor Safety System Development	8	5	90	530
7.	SODIUM TECHNOLOGY				
7.1	Liquid Metal Purification System	14	7	320	1,050
7.2	Cover and Hot Cell Gas Purification Systems	13	7	175	865
7.3	Development of Non-Gaseous Fission Product and Active Corrosion Product Management Schemes	14	10	260	1,050
7.4	Sodium Quality Control	5	5	150	460
7.5	Sodium Sampling and Analysis	5	5	75	385
8.	SAFETY AND ENGINEERED SAFEGUARDS				
8.1	Radiation Shielding Development	16	8	200	1,030
8.2	Blast Shielding Development	10	5	240	760
8.3	Containment Design Basis Development	24	10	380	1,590
8.4	Protection Against Loss of Coolant	12	8	280	950
8.5	Protection Against Secondary Criticality	12	6	260	990
8.6	Radiological Protection	8	3	30	430

3. COST OF FACILITIES

Existing and planned Westinghouse experimental facilities are part of Westinghouse capital investments. Many are cited in the Facility Requirements section of the program descriptions of Section III and are described in Appendix A. These facilities have been or will be capitalized and their depreciation becomes part of overhead costs applied to direct salary expenditures for engineers, scientists, draftsmen, and technicians. The costs of these facilities are included in the manpower cost estimates (Table 2-1).

Cost estimates for major AEC facilities which are existing, or are currently being designed or constructed, are not provided here because government records and estimates of the constructing agency would be more accurate than could be developed under the scope of this study. For this reason, projected in Section III are those needed major government facilities which are not currently identified in active design and construction programs. Rough estimates of the capital costs for these facilities are listed in Table 3-1.

TABLE 3-1
MAJOR TEST FACILITIES

1. Fuel Rod Fabrication Facility

Purpose: To fabricate (U,Pu)C fuel assemblies with an output of ~200 fuel rods per week. (This is the expansion of a facility such as the W Plutonium Lab.)

R&D Task: Evaluation of Fuel Fabrication Processes (3.1.A)

Date Required: 1971

Estimated Cost: \$0.5-1.0 million

2. Post-Irradiation Examination Facility

Purpose: To provide for post-irradiation examination of fuel rods and assemblies. (The only need is for α shielding capability, which could be added to an existing facility such as at WARD.)

R&D Tasks: • Fast Flux Irradiations (3.1.F)
 • Thermal Flux Irradiations (3.1.G)
 • Transient Rod Testing (3.1.H)

Date Required: 1971

Estimated Cost: ~\$5.0 million

3. Transient Test Facility

Purpose: To provide the capability of testing fuel rods and sub-assemblies under transient conditions by superimposing a power insertion over normal operating power.

R&D Tasks: • Transient Rod Testing (3.1.H)
 • Destructive Tests for Determination of Energy Releases (4.4.H)

TABLE 3-1 (Continued)

- Evaluation of Canned and Uncanned Fuel Assemblies with Regard to Blow Blockage and Void Propagation (4.4.C)
- Determination of Effects of Accidents in Modular and Non-Modular Cores (4.11.C)
- Fuel Collapse Studies (8.3.E) (8.5.A)
- Fuel Motion Studies (8.5.C)
- Cooling Methods Development (8.5.D)

Date Required: 1971

Estimated Cost: \$100-150 million

4. Reactor Flow Model Test Facility

Purpose: To test scale models of reactor cores and vessel internals in a sodium environment (~20,000 gpm); to study pipe rupture, multi-loop flow characteristics, sodium hammer effects and siphon breaker behavior.

- R&D Tasks:
- Studies of Coolant Flow Distribution in Reactor Vessel (4.9.A)
 - Development of Coolant Siphon Breaker (4.9.B)
 - Determination of the Effect of Vortex Suppressor on Pressure Drop (4.9.D)
 - Determination of Pipe Failure Modes (8.4.A)
 - Double Containment Piping Studies (8.4.B)
 - Formulation of Primary Loop Rupture Models (8.4.C)
 - Sodium Hammer Investigations (8.4.D)
 - Thermal-Hydraulic Model Studies (8.4.E)
 - Siphon Breaker Studies (8.4.F)

Date Required: 1970

Estimated Cost: \$2.0-3.0 million

TABLE 3-1 (Continued)

5. Non-Destructive U/Pu Assay Facility

Purpose: To non-destructively determine U/Pu ratio utilizing x-ray spectroscopy and/or a ~150 KeV generator of monoenergetic neutrons.

R&D Task: Development of Methods for Sensing U/Pu Ratio (4.10.D)

Date Required: 1974

Estimated Cost: \$0.1-0.2 million

6. On-Line Computer Facility

Purpose: To perform continuous fuel management for the Demonstration Plant.

R&D Task: Test of Fuel Management System in Demonstration Plant (4.10.E)

Date Required: 1976

Estimated Cost: \$1 million

7. Sodium Combustion Facility

Purpose: To obtain experimental data pertaining to sodium burning; to study sodium spray fire and sodium aerosol; to evaluate engineered safeguards.

R&D Tasks: • Sodium Combustion Studies (8.3.A)
• Determination of Sodium Spray Fire Data (8.3.B)
• Aerosol Spread of Radioactivity Studies (8.3.C)
• Engineering Safeguards Studies (8.3.D)
• Aerosol Settling Mechanism (8.6.A)
• Aerosol Settling Device Development (8.6.B)

Date Required: 1971

Estimated Cost: \$2-3 million

4. COST OF THE DEMONSTRATION PLANT PROGRAM

The Demonstration Plant Program, as described in Section III-2.2 of this report, is divided into five phases:

- Pre-Construction R&D Program
- Title I and Title II Plant Design
- Plant Construction
- Post-Construction R&D Program
- Plant Operation Analysis

The estimated cost of the entire program will be between 150-200 million dollars. This figure includes the capital cost of the Demonstration Plant, but it excludes the expenses for major test facilities as outlined in Section IV, 3. This cost can be divided into three major categories, namely:

- a. Items funded by the AEC, an Electric Utility or group of utilities, and Westinghouse
- b. Items funded by an Electric Utility or a group of utilities
- c. Items funded by Westinghouse alone.

The items which will be funded by the combination of the AEC, the Electric Utility industry, and Westinghouse resources will cost an estimated 100 million dollars, and will include the items listed on pages 2-146 and 147 in Volume 2 of the LMFBR Program Plan. These items are:

1. R&D, both pre -and post-operational
2. First-of-a-kind engineering
3. First-of-a-kind fabrication, including the associated work needed to up-grade standards and to proof test equipment
4. Supply of early cores (including financing of related fuel-cycle cost)
5. Licensing
6. Reactor post-operational analysis
7. Design and development of special instrumentation

The items funded by the utilities themselves include the capital cost items of the Demonstration Plant. The estimate for this is 60-90 million dollars, excluding interest during construction.

Westinghouse is funding ~20 million dollars in facilities at the Waltz Mill site. Some of these facilities are described in the Appendix of this report. They consist of the laboratories, and offices and equipment needed to perform much of the pre-construction R&D program for the Demonstration Plant.

SECTION V

VALUE ANALYSIS

1. INTRODUCTION

1.1 Purpose

The utility industry has repeatedly stated through word and action that, in the final analysis, the choice of one source of electric power generation over another will rest not upon the technical superiority, but upon the economic competitiveness of the power source. This economic challenge has been accepted by the USAEC and the nuclear industry and will influence the R&D objectives for development of the LMFBR and other future concepts. It is the intent of this section of the report to examine the proposed R&D plans and to evaluate the consequences of the success or failure of these plans against the economic yardstick of LMFBR projected power cost.

Specifically, the scope of the work is:

A quantitative assessment of the effect on the Reference Concept Design, and an estimate of the incremental penalty in the cost of energies produced by the Reference Concept Power Plant, which would result from non-attainment or only partial attainment of the research and development objective for the item. The contractor may indicate alternate paths to satisfy an objective if desired.

1.2 Scope, Depth and Significance of Study

A total of two-hundred and eight R&D tasks have been proposed in Task IV, with many having multiple objectives which affect plant cost. It should be recognized that neither schedule nor level of program support permitted an economic assessment of the merits of all of these tasks. Consequently, attention was focused on those areas exhibiting the most potential for influencing the power cost. However, it must be cautioned that the value analysis technique is only part of the overall process in evaluating the merits of an R&D plan, and the absence of an R&D task evaluation in this section should not imply that Westinghouse does not consider it important in the LMFBR development plan.

Some R&D areas are not amenable to value analysis, as their objectives and end results are not readily related to plant economics; e.g., calculational methods development, basic materials research, instrumentation and control, and sodium technology. Nevertheless, the contribution of these R&D areas remains critical to the successful development of the LMFBR.

Another limitation of value analysis is that R&D program failures can provide significant cost savings by eliminating weak design or material features from the final plant design, but it is difficult to predict these savings through value analysis.

The basic cost input into the value analysis study has been derived primarily from the parametric studies conducted under Task III where the sensitivity of power cost to variations of key plant parameters from optimum values was quantitatively calculated. Each R&D task was evaluated and assigned a probability of attaining its important objectives. This assessment is largely a matter of judgment and, in some cases, speculation. The combined consideration of cost sensitivity and success probability provides some measure of the value of the R&D program.

1.3 Cost Benefit Considerations

The minimization of LMFBR power cost through R&D is a noteworthy objective, provided the resources expended through the R&D programs do not exceed the returnable economic benefits. Several broad scope studies have proven that the overall development of the LMFBR can be justified on the basis of potential economic returns to the national consumers of electrical power.

A similar demonstration of benefits on a more refined scale should be made for the individual R&D tasks in the LMFBR program. Although not part of the scope for this study, a few simple calculations were made to determine the magnitude of power cost savings for the LMFBR that would provide a reasonable cost benefit.

The Fast Breeder Reactor Report of the Edison Electric Institute utilized a projected FBR market penetration in their studies which assumed the introduction of economic 1000 MWe fast breeder reactors in 1980 at the following rate: four in the first two years, eight in the next two years, 16 in the next two years, etc., until 1989 after which all new nuclear additions are assumed to be fast breeder reactors. Relating this growth rate to cumulative FBR capacity, results in 2000 MWe in 1980, 28,000 MWe in 1985, and 160,000 MWe by the end of 1990.

The 1970 present worth economic return (using 7 percent cost of money and LF of 0.8) for an assumed savings in power cost of 0.1 mil/kwhr for the above fast breeder reactor installations has been calculated as \$116 million over the period 1980 through 1990. While, admittedly, the FBR market projection is still a matter of conjecture, a potential power cost savings of 0.1 mil/kwhr can justify substantially-sized R&D programs. As a further illustration, a market capture rate of one-tenth as large or a 0.01 mil/kwhr power cost savings for the above EEI case would still represent a \$12 million present worth return.

1.4 R&D Programs of General Significance to Power Cost

A brief analysis of the LMFBR power cost is warranted to ascertain the distribution of this cost among the reactor and plant systems. This distribution can be utilized to infer potential technical areas where R&D can produce potential cost reduction.

The bus bar power cost for the LMFBR single module design has been calculated as 4.109 mils/kwhr and is distributed among the usual accounts as follows:

Fixed cost on capital investment	3.190 mils/kwhr
Fuel cycle	0.519 mils/kwhr
Operations and maintenance	0.400 mils/kwhr
Total	<u>4.109 mils/kwhr</u>

Basis: Load Factor 0.8

It is apparent from comparison with typical cost distributions for the LWR that the LMFBR shows higher revenue requirements for capital investment, but these are compensated for by considerably lower fuel cycle revenue requirements. It will be shown in a subsequent section that the lower fuel cycle cost is mainly attributed to the much higher burnups achievable in the fast breeder reactor. Because failure to meet projected burnups would have serious implications to the competitiveness of the LMFBR, fuel performance studies require and have obtained the highest priorities in the LMFBR program.

A breakdown of the fuel cycle cost shows the following increments of cost:

	<u>Percent of Total</u>	<u>Mils/kwhr</u>
Fabrication	78	0.405
Plutonium Credit	(-84)	(-0.436)
Inventory	74	0.384
Shipping Cost	8	0.042
Reprocessing Cost	<u>24</u>	<u>0.124</u>
Total	100	0.519

Next to burnup, the cost of plutonium fuel fabrication has the greatest impact upon overall fuel cycle cost. As in the case of burnup, the current state-of-the-art for plutonium fabrication must be substantially improved by R&D to match the projected values used in the above estimates. Inventory costs, which consist of carrying charges for the investment in fuel (fissile value and fabrication) contribute substantially to fuel cycle cost, but are offset by plutonium credit received for bred fuel. It should be mentioned that plutonium credit is currently based upon a somewhat arbitrary plutonium value established by the AEC. During the initial period of growth for the LMFBR, the plutonium value most likely will be determined by its value to the LWR operator. This value will be computed by consideration of the equivalent uranium-235 value less any differentials in fabrication costs between uranium and plutonium. Thus, reduction in fabrication costs for plutonium would reduce cash flow requirements for the LMFBR operator and enhance the value of bred plutonium.

Reprocessing costs were essentially fixed by the basic ground rules of the study. However, the importance of these costs to the fuel cycle should not be de-emphasized. Actually, considerable uncertainties exist in these costs as a new head-end treatment is required for sodium-bonded carbide fuels; the effects of high burnup fuel upon radiolytic degradation of the organic solvent and throughput rates for FBR enrichments have not been established; and finally, commercial experience with reprocessing plants is very limited. Although not discussed further, an R&D program which would delve into these areas is worthy of serious attention.

The reference nominal 1000MWe LMFBR Plant has been priced at 180 million dollars or a corresponding unit price of \$173 per kw. Compared to recent quoted prices for the LWR's of about \$150/kw, the capital cost of the LMFBR is approximately 15 percent higher. This conclusion is in agreement with the recent "Fast Breeder Reactor Study" by the Edison Electric Institute which estimated the capital cost of fast breeder reactors to be 20 percent more than LWR by 1985, 15 percent by 1990, 5 percent by the year 2000.

The cost associated with the major systems of the plant is shown by percentage and unit price as follows:

	<u>Percent of Total</u>	<u>Unit Price \$/Kw</u>
Reactor system	38.6	66.8
Turbine plant	35.6	61.6
Service systems	<u>25.8</u>	<u>44.6</u>
Total	100.0	173.0

As the current state-of-technology is well advanced for turbine plant systems, little cost improvement can be expected in this area even with the continued R&D expended by the manufacturers. The major benefits derived by the R&D will be accrued in the areas of reliability and maintainability of turbine plant equipment. State and federal legislation affecting thermal rejection into rivers and costal waters could result in increased capital and operating costs for the turbine plant system.

The reactor system designs have benefitted immensely from the technology developed for the LWR and while R&D is required to develop the reliability of components in sodium, only modest cost savings will most likely be achieved in this area. Improvements of 10-15 percent in installed cost or \$7 to \$10 per kw would be a fair objective.

The service systems represent an area where considerable improvement in cost is feasible. Conversely, the requirements and risks for R&D investments are also the highest. Currently, our cost estimates of the service systems are limited by our abilities to project beyond the present level of knowledge and therefore as a rule, probably tend to be conservative. Costly service systems where R&D may achieve savings are nuclear fuel handling, auxiliary cooling systems, plant gas systems, hot cell environmental system, and the reactor coolant receiving, supply and treatment system. Improvements of 20-50 percent in cost appear to be reasonable R&D targets and could produce a unit price reduction of \$9 to \$22 per kw for the plant.

Return on investment, depreciation, interest, insurance and administrative costs for the \$180 million plant investment generate a fixed revenue requirement for the plant which relates to a capital cost increment at an 80 percent load factor of 3.190 mils/kwhr. The importance of high availability for the plant is clear, e.g. a decrease to 70 percent load would raise the capital cost increment to 3.646 mil/kwhr, an increase of 0.46 mil/kwhr.

2. RESULTS AND DISCUSSION

2.1 General

The assessment of the economic consequences to the power cost resulting from the success or failure of an R&D program requires as input the following information:

- a. Definition of the objectives for the R&D task
- b. Correlation of the objectives to tangible reactor or plant technical parameters that directly or indirectly affect power cost
- c. Delineation of the effective range of the technical parameter values that can be influenced by the R&D plan
- d. Calculation of power cost sensitivity to the range of technical parameter values
- e. Assessment of probability of attaining desired parameter values

The definitions of the R&D objectives are provided in the R&D program description section of this Task IV report. Also, the summary tables accompanying the R&D descriptions have delineated functional specifications in terms of current and projected states-of-technology and serve to partially fulfill the informational requirements of items b and c. Sensitivity of cost to key plant parameters is obtained from the Task III parametric studies whenever possible. The final informational need, the assessment of R&D success probabilities, is the most difficult to obtain and is based primarily on the qualitative judgment of experienced R&D personnel.

A set of definitions of the probability of R&D success was established for this analysis. The programs are ranked as Excellent, Good, and Fair, based upon the following rationale:

1. Excellent - High probability of success with no compromise in objectives
2. Good - May require compromise in objectives but represents substantial improvement over current technology
3. Fair - May require substantial compromise in objectives resulting in little improvement over current technology

Similarly, program priorities have been assigned a 1, 2, or 3 rating utilizing the qualitative definitions established on page 1-42 of Volume 1 of the LMFBR Program Plan, WASH-1101.

The results of the value analysis are summarized in Table 2.1-1 in a task format corresponding to the R&D program as presented in this report. The column entitled "R&D uncertainty or sensitivity of Power Cost Increment" represents the range in fuel cycle cost, capital cost, or total power cost that is influenced by the degree of attainment of the R&D objective.

TABLE 2.1-1
VALUE ANALYSIS

Task	Title	Objective	Design Parameters or Areas Affected by R&D Results	Primary Effects of Power Cost	Power Cost Increment (R&D Uncer- tainty or Sensitivity)	Proba- bility of Attaining R&D Objective	Priority Assigned R&D Task
3.	<u>FUELS AND MATERIALS</u>				<u>mils/kwhr</u>		
3.1.A	Fuel Fabrication Processes	Develop Economical Material Fabrication Processes	Rod Fabrication Process	Fuel Cycle	0.25	Good	1
3.1.B	Fuel Properties Determination	Characterize Physical and Mechanical Properties of Mixed Carbide Fuels	Burnup Limit Fuel Centerline Temp. Thermal Conductivity	Fuel Cycle Fuel Cycle Fuel Cycle	0.32 0.08 0.08	Excellent	1
3.1.C	Cladding Properties	Characterize Cladding Properties (Unirradiated)	Cladding Surface Temp.	Fuel Cycle and Capital	0.09	Excellent	1
			Burnup Limit	Fuel Cycle	0.32		
			Rod Design	Fuel Cycle	0.05		
3.1.D	Fuel Cladding Compatibility	Fuel/Bond/Cladding Compati- bility	Burnup Limit Rod Design and Specifi- cations	Fuel Cycle Fuel Cycle	0.32 0.03	Good	1
			Stoichiometry Control	Fuel Cycle	0.05		
3.1.E	Fuel Rod Fabrication Procedures	Develop Bonding and Weld Techni- ques	Rod Fabrication Cost	Fuel Cycle	0.01	Good	1
3.1.F	Fast Flux Rod Irradiation	Irradiation Performance of Fuel and Cladding	Stoichiometry	Fuel Cycle	0.05	Fair	1
3.1.G	Thermal Flux Irradiation		Burnup Limits	Fuel Cycle	0.32	Fair	1
3.1.H	Transient Rod Testing		Temp. Limitations on Fuel	Fuel Cycle	0.08	Fair	2
3.1.I	Cladding Properties (Irradi- ation)		Cladding Surface Temp.	Fuel Cycle and Capital	0.09	Good	1

TABLE 2.1-1 (Continued)

Task	Title	Objective	Design Parameters or Areas Affected by R&D Results	Primary Effects of Power Cost	Power Cost Increment (R&D Uncer- tainty or Sensitivity)	Proba- bility of Attaining R&D Objective	Priority Assigned R&D Task
					<u>mils/kwhr</u>		
3.2	Core Structural Materials	Develop Materials Properties for Fuel Assembly Grids, Reactor Vessel and Vessel Internals	Grid, Strongback Design Vessel and Internals Design Reliability	Fuel Cycle Capital Power	0.08 Nil 0.01	Good	1
3.3	Steam Generator Materials	Acquisition of Properties, Corrosion and Mass Transfer Data in Sodium and Water	Reliability/Plant Avail- ability Unit Size (Single versus Multiple units) Engineering Design Factors ^b	Power Capital Capital	0.23 0.05 Nil	Good	1
3.4	HX Materials	Acquisition of Properties, Corrosion and Mass Transfer Data in Sodium	Reliability/Plant Avail- ability Engineering Design Factors ^b	Power Capital	Nil Nil	Excellent	2
3.5	Large Sodium Pump Materials	Acquisition of Properties, Corrosion, Erosion, Mass Transfer Data for Pump Materials	Reliability Engineering Design Factors ^b	Power Capital	0.04 Nil	Excellent	1
3.6	Check Valve Materials	Develop Materials and Determination (Properties and Wear Characteristics)	Safety	Capital	Nil ^a	Excellent	2
4.	<u>REACTOR DEVELOPMENT</u>						
4.1	Studies of Temp. Coef. for (U,Pu)C	Determine Temp. Coef. of Reactivity	Core Neutronics	Fuel Cycle	c	Fair	2
4.2	Develop Na Fuel Bond	Determine Behavior of Na Bond Subject to High Temp. Gradients, Fission Products, and Impurities	Fuel Rod Life Fabrication - Bond Techniques	Fuel Cycle Fuel Cycle	0.32 0.01	Fair	1

TABLE 2.1-1 (Continued)

Task	Title	Objective	Design Parameters or Areas Affected by R&D Results	Primary Effects of Power Cost	Power Cost Increment (R&D Uncer- tainty or Sensitivity)	Proba- bility of Attaining R&D Objective	Priority Assigned R&D Task
					<u>mils/kwhr</u>		
4.3	Determine Hydraulic Effects on Fuel Cladding	Determine Cladding Erosion, Grid Fretting Resulting from Hydraulic Forces	Assembly Design Number and Type Grids	Fuel Cycle Fuel Cycle	Nil ^a 0.08	Excellent	1
4.4	Studies of Na Coolant Behavior	Investigate Na Boiling, Void Generation, Hot Channel Factors	Hot Channel Factor	Capital	0.1	Fair	1
4.5	Develop Fuel Assembly	Studies Relating to Thermal, Hydraulic, and Mechanical Evaluation of Fuel Assembly Design	Type and Number of Grids Velocity Limitations Hot Channel Factor	Fuel Cycle Power Capital	0.08 0.04 0.1	Excellent	1
4.6	Develop Radial Blanket	Studies Relating to Flow Dis- tribution, Hydraulics Character- istics, Orificing For Blanket Assemblies	Blanket Fabrication Cost	Fuel Cycle	0.03	Excellent	1
4.7	Develop Control Rods and Drives	Study Na and FP Effects on Control Rod Mechanisms and Conduct Worth Measurements	Control Worth Materials and Mechanical Design	Fuel Cycle Capital	0.05 0.04	Excellent	1
4.8	Studies of Modular Geometry	Critical Experiments with Modular Geometry to Determine Neutronic Coupling, Feed Back Constants, Void and Control Worth	Na Void Criterion - Core Concept	Fuel Cycle	0.25	Good	2
4.9	Develop Reactor Vessel Internals	Hydraulic Studies on Internals Development of Siphon Breaker	Internals Design and Reliability Siphon Breaker (Safety)	Capital Power	0.01 0.04	Good	1
4.10	Develop Computer-Assisted Fuel Management System	Evaluate Fuel Management Methods Assess Canned versus Uncanned Assemblies and Develop In-Core Sensors	Fuel Management	Fuel Cycle	0.1	Good	2
4.11	Develop a Non-Modular Core	Studies of Void Propagation, Accident Effects	Safety	Fuel Cycle	0.25	Fair	2

TABLE 2.1-1 (Continued)

Task	Title	Objective	Design Parameters or Areas Affected by R&D Results	Primary Effects of Power Cost	Power Cost Increment (R&D Uncer- tainty or Sensitivity)	Proba- bility of Attaining R&D Objective	Priority Assigned R&D Task
5.	<u>PLANT SYSTEMS AND COMPONENTS</u>				mils/kwhr		
5.1	Develop Reactor Vessel	Develop Design Codes, Inspection Techniques, and Closure seals	Vessel Design Techniques	Capital	Nil	Excellent	1
5.2	Develop Main Transport Piping	Develop Design Methods, Pipe Support, Field Fabrication Methods, Inspection and Maintenance Procedures for Piping	Piping System Design Maintenance Procedure	Capital O and M	Nil 0.05	Excellent	1
5.3	Check Valve Development	Develop Large Check Valve for Na Service	Safety Feature	Power	0.04	Good	1
5.4	Develop Large Sodium Pump	Develop Shaft Seals, Bearings, Mechanical and Hydraulic Design, and Test of Large Na Pumps	Pump Size/Number Loops Reliability	Capital Power	0.1 0.04	Excellent	1
5.5	Develop Sodium-to-Sodium Heat Exchanger	Develop Weldments, QC and Inspection Methods, Maintenance Procedures, and Design Data for Na-to-Na HX	Maintainability Reliability	Power Power	Nil Nil	Excellent	1
5.6	Develop Steam Generator	Thermal and Hydraulic Studies; Develop Fabrication and QC Procedures; Sodium/H ₂ O Reaction Studies; Prototype Testing	Reliability and Maintainability Unit Size/Number of Units	Capital Capital	0.23 0.05	Good	1
5.7	Develop Fuel Handling and Maintenance Systems	Develop Components and Sub-systems For Hot Cell Refueling System	Refueling Times/Plant Availability Refueling System Design	Capital Capital	0.16 0.03	Excellent	1
6.	<u>INSTRUMENTATION AND CONTROL</u>	Develop Reactor Plant Instrumentation and Control Sensors and Systems	Control System Concept Fuel Management Component Failure Detection	} Fuel Cycle, Capital	c c	Good to Excellent	1

TABLE 2.1-1 (Continued)

Task	Title	Objective	Design Parameters or Areas Affected by R&D Results	Primary Effects of Power Cost	Power Cost Increment (R&D Uncer- tainty or Sensitivity)	Proba- bility of Attaining R&D Objective	Priority Assigned R&D Task
7.	<u>SODIUM TECHNOLOGY</u>				<u>mils/kwhr</u>		
7.1 and 7.3	Purification System Development, Non-Gaseous Fission Product Management	Develop Processes and Equipment For Control of Na Chemistry	Coolant Purity Levels (Influence Materials Con- siderations, e.g. Cor- rosion, mass transfer rates; shielding)	Capital, O and M	0.03 0.02	Excellent	1
7.2	Cover and Hot Cell Gas Purifi- cation Development	Develop Processes and Equipment For Maintenance of Reactor and Refueling Cell Atmosphere Purity	Fission Gas Removal and Control Sodium Impurity Levels Hot Cell Design	Capital, O and M	0.07 c	Excellent	1
7.4	Sodium Quality Control	Develop In-Line Monitors Develop Analytical Methods and Techniques	Component Failure Detection Coolant Chemistry Control	Power Cost, O and M	c c	Good	1-2
8.	<u>SAFETY AND ENGINEERED SAF- GUARDS</u>	Analyze Potential Accidents and Provide Engineered Safe- guards	Number and Extent of Engineered Safeguards	Capital	c	Good	1

a. Not determined but probably Nil

b. Engineering design - selection of materials, stress loading, dimensional specifications, heat treatment, hot or cold-worked, etc.

c. Not determined, but significant

2.2 Fuels and Materials

Numerous economic studies of projected LWR power costs for the 1980's have concluded that the LMFBR development must aim for a fuel cycle cost in the range of 0.5 to 0.7 to assure a sound competitive position for the LMFBR. The two fuel parameters virtually dictating fuel cycle cost are burnup and fuel fabrication cost. The interrelationship between fabrication cost, burnup, and fuel cycle cost is depicted in Figure 2.2-1. Reference design targets are 100,000 MWD/MTU average burnup, and \$264/Kg (PU + U) fabrication cost which yield a fuel cycle cost of 0.52 mil/kwhr.

At the present, extrapolations of these parameters from current-state-of-technology are beset with considerable uncertainty.

Burnup limitations upon the mixed plutonium-carbide fuel will be fixed by the combined output of several R&D tasks which study the following:

- a. Fuel swelling and fission gas release
- b. Compatibility of cladding and bond with carbide fuel
- c. Effects of irradiation exposure on cladding
- d. Mass transfer and corrosion effects upon cladding
- e. Erosion and grid fretting wear of fuel rods under hydraulic forces

The minimum economically acceptable average burnup limit is 50,000 MWD/MTU. Current capsule tests in GETR should establish the feasibility of attaining this level of irradiation. Assuming further incentive to proceed with carbide development, subsequent fuel cycle cost savings by extending the average burnup to 100,000 MWD/MTU would be 0.28 mils/kwhr and 0.32 mil/kwhr for 150,000 MWD/MTU.

The factor contributing the greatest uncertainty to the cost of fabrication is the preparation of the carbide feed material. The current method of arc melting to produce carbide feed material is costly, and not conducive to volume production and close control of stoichiometry. Utilization of this method in a first generation commercial fabrication plant would yield fabrication costs of around \$550/kg, a value that is economically unacceptable. Thus, substantial priority should be given the R&D task to develop the carbothermic method of feed material production, as this process shows

Figure 2.2-1. Fuel Cycle Cost as a Function of Burnup and Fabrication Cost

promise for savings with volume production and is projected to give \$264/Kg costs with the same size plant. The potential for power cost reduction is around 0.25 mil/kwhr.

The fabrication cost is roughly split 65 percent for rod manufacture and 35 percent for assembly costs. The assembly costs are currently high because fabrication techniques for the new grid designs to accommodate the tight rod lattices and the actual number of grids to eliminate fuel rod bowing, vibration, and fretting still require resolution. It is estimated that 0.15 mil/kwhr of the fuel cycle cost is associated with the uncertainties in the assembly stage.

The results of the fuels development programs can impose other limitations on parameters such as thermal conductivity, cladding thickness, fuel centerline temperature, cladding surface temperature, and type of fuel rod bond. In general, the power cost implications are less than 0.1 mil/kwhr but still significant. For example, cladding compatibility considerations could require extreme close stoichiometry control and/or Cr or Fe additions. The impact on the fabrication cost would be increased process reject rates and consequently, increased material recycle rates. In the case of Cr additions, the reject material is extremely difficult to reprocess and fuel cycle cost could be affected by as much as 0.05 mil/kwhr. The sensitivity of power cost to some of the above parameters is depicted in Figures 2.2-2 and 2.2-3.

The R&D programs for the development of materials for the core structures, steam generator, intermediate heat exchanger, and large sodium pumps will affect power costs by influencing materials choices, design parameters, fabrication techniques and, perhaps in some cases, maximum unit sizes. In most instances, the R&D tasks will gather and confirm data on physical and chemical properties of materials for the designer. The direct impact on component costs from these data is usually minor (less than 0.01 mil/kwhr), but failure to obtain data could result in reduced component lifetime and significant replacement cost. For example, data of mechanical properties obtained for structural materials in sodium most likely would not affect the cost of the core internals. Failure of an over-stressed structural member, however, could cause extensive core internals damage. Replacement and/or

Figure 2.2-2. Sensitivity of Power Cost to Fuels R & D Target Objectives

Figure 2.2-3. Sensitivity of Power Cost to Fuels R & D Target Objectives

repair of core internals even once during plant life would result in an additional 0.01 mil/kwhr power cost.

Similarly, results from materials studies could improve frequency of repair factors for components. With the high capital cost of the LMFBR, unscheduled outages of the plant cause substantial increases in revenue requirements. An unscheduled annual outage for a single primary pump could result in 0.03 mil/kwhr increased power cost. In the case of the steam generator a tube failure resulting in only local damage could result in 0.16 mil/kwhr additional revenue requirement for the year of occurrence. Catastrophic failure of a steam generator requiring replacement in addition to the costs of outage could increase annual revenue requirements by 0.07 mil/kwhr. These cost values, while admittedly arbitrary, point out the potential gains that the materials studies can indirectly contribute.

2.3 Reactor Development

The reactor development programs encompass R&D studies relating to the physics, thermal and hydraulic, and mechanical design of the core.

Major emphasis is placed upon the mechanical and thermal/hydraulic aspects of fuel assembly design, including:

- a. Na bond behavior under various operating and purity conditions
- b. Thermal and hydraulic performance
- c. Erosion of cladding under high velocity conditions

Power cost improvements potentially accrue from extension of fuel assembly lifetime, and assembly design features. The implications of burnup have already been discussed in the previous section. Fuel assembly design features to be studied will include grid design, strongbacks, holddown mechanisms, nozzles, and number of grids required to prevent fuel rod bowing and vibration. Because of the tight fuel rod lattices required for optimum fuel volume fraction, the current grid and grid support designs are expensive. Currently assembly costs would constitute \$131/Kg of the fabrication cost. With contemplated improvements in design, fabrication techniques, and number of grids per assembly, a reduction of \$40/Kg or 0.08 mil/kwhr in fuel cycle cost is feasible.

The performance studies of the sodium bond under high temperature transients and in the presence of impurities and fission products will determine the bond loading and inspection technique necessary. The final chemical and physical specifications generated for the bond could influence the fuel cycle cost by 0.01 m/kwhr.

At present, hot channel factors penalize the power cost by 0.1 mil/kwhr. Recovery of a large fraction of this penalty could be permitted by improvements in engineering factors, development of core orificing, and hydraulic mixing studies.

In theory, the fuel cycle cost is very sensitive to the choice of core physics parameters, such as fuel volume fraction, control rod worths, coefficients of reactivity, neutronics data, and flux/power distributions. However, this is an area where the complexity of the subject permitted only a superficial

analysis. Also with the current state-of-the-art, the practical parametric ranges of interest are considerably narrowed by thermal, hydraulic, and mechanical constraints. Thus, as an example, the incentive to increase fuel volume fraction, while desirable, is restricted by practical considerations of grid design and sodium velocity limitations.

Several core physics areas present noteworthy R&D objectives. At present, there is a 20 percent uncertainty in the control worth of proposed control rod materials and designs. This uncertainty in control results in an unnecessary reduction in fuel volume fraction and a loss of 0.05 mil/kwhr. The proposed control rod development program has a high probability of success for considerably reducing this uncertainty. In the same way, uncertainty in the nuclear data, mainly α , the ratio of capture to fission in Pu-239 below 15 Kev, contributes an error band of ± 0.03 mil/kwhr in the fuel cycle.

Experience with LWR's, particularly the Yankee Core Evaluation Program, has shown that fuel management can provide substantial savings to the fuel cycle. Admittedly, while potentialities of fuel management in the LMFBR are limited by the reduced number of fuel assemblies, it is estimated that savings as great as 0.1 mil/kwhr can still be a reasonable R&D target objective for fuel management studies.

A very important core physics consideration that will require detailed and careful study is the criterion for sodium voiding. During the Task II phase of the study, calculations were made for three basic core configurations: a seven module reactor, a four module reactor, and a one module reactor. In principle, each configuration should meet the criterion for sodium voiding, i.e., zero or negative reactivity for sodium voiding of the core. However, in practice the economic penalties to fuel cycle cost are substantial as demonstrated in Figure 2.3-1. The fuel cycle costs computed for the two optimized designs in this study are 0.52 mil/kwhr for a single module core and 0.78 mil/kwhr for the four module core. The difference is slightly more than a quarter of a mil per kwhr. The single module core, of more conventional design, has a slightly positive reactivity upon sodium voiding, while the four module core design adheres to the negative void criterion.

Figure 2.3-1. Fuel Cycle Cost as a Function of L/D Module Ratio

A valid questioning of the need for a negative void criterion, the introduction of other control problems with multi-module cores, and the economic penalties associated with core designs incorporating negative void criterion suggest that the single module core of unspoiled geometry be adopted as the reference design for further pursuit. However, an R&D program of intermediate priority is recommended for further study of multi-modular concepts, to establish feasibility and to explore possibilities of lowering economic penalty.

2.4 Plant Systems and Components

The programs included in this category involve the development of the following systems and components: reactor vessel, main transport piping, intermediate heat exchanger, steam generator, fuel handling systems, and instrumentation and control.

The R&D programs of plant systems and components are difficult to value analyze because the developmental results that lead to improved designs and operational reliability are often unpredictable. For this type of development program, an assessment of the uncertainties in cost estimates for components and system provides a clue as to the direction and magnitude of potential developmental savings. Prior operating experience, although limited, provides an indication of the potential reliabilities of the components and systems.

The above approach was utilized to determine that the following systems and/or components have attained a reasonably sound state-of-the-art and should present no unusual reliability problems: 1) reactor vessel, 2) main transport loops, and 3) sodium-to-sodium heat exchangers. The uncertainty of the cost estimates for each of these systems is about 10 to 15 percent. The net effect on the total power cost is minor (less than 0.01 mil/kwhr).

The major problem confronting the sodium-heated steam generator designer is that tube failure rates normally acceptable in air/water or water/water applications are intolerable for sodium application because of the potential catastrophic consequences of the H_2O/Na reaction and the extreme difficulties and hazards associated with repair. With the current state-of-technology, it is not inconceivable that units would have to be replaced or extensively repaired at least once during plant life. Thus, as a minimum, the R&D incentive for improving reliability is the cost of replacement, or 0.07 mil/kwhr additional power cost. With initial plants and also when desired levels of reliability are not attained, packaging the required heat transfer capacity into multiple units must be considered. Multiple units would increase capital costs by approximately 0.05 mils/kwhr.

Similar questions of reliability exist with large sodium pumps, particularly as size is increased, although consequences of failure are not so severe. Also, the R&D programs may conclude a practical limit to pump size. In any event, the plant designer will have to weigh the reliability factors of various pump sizes against multiple coolant loops to arrive at an optimum arrangement. The effect upon power cost of increasing the number of coolant loops from two to four would be 0.1 mils/kwhr.

In addition to representing a large capital investment, the fuel handling system can influence the plant availability factor to a substantial degree. For example, reduction of refueling times by two weeks would mean a power cost savings of 0.16 mil/kwhr. Because the refueling system is still in the conceptual design stage, cost estimates for the system and subsystems are subject to uncertainties as high as 50 percent or 0.03 mil/kwhr power cost.

The impact of instrumentation and control development upon costs was not analyzed. The availability of suitable sensors could, however, influence greatly the success of programs such as fuel management, core orificing, failure detection of fuel assemblies, steam generators, and other components.

2.5 Sodium Technology and Safety

Present sodium technology and safety aspects of the plant design are largely conceptual. Systems have been described and included in the reference design that will satisfy postulated safety requirements and maintain the stated levels of purity control for liquid metal and gas environment. Extensive R&D is required however, not only to search for better processes of purity control, but also to establish purity specifications that are based upon actual materials experience and operational demands. To a large extent engineered safeguards have been based upon precedents established by LWR experience. The entire area of LMFBR safety must be studied, and a set of safety criteria developed that is unique to the LMFBR.

It was not possible to perform a value analysis of the above areas. However, a qualitative examination shows that significant gains in power cost could be accrued from R&D programs. Collectively, the uncertainties in the cost estimates for the sodium and gas environmental service systems account for 0.1 mil/kwhr in power cost.

APPENDIX A

WARD Facilities

Many of the R&D programs in Section III of this report require facilities of the type either presently in operation at the Westinghouse Advanced Reactors Division or in the planning, or building stages. These facilities are described in this Appendix to indicate their design features and operating characteristics where appropriate. Reference to these descriptions is made in the individual R&D program where a facility of a type existing at ARD is required.

The facilities of the division can be classified into functional groupings corresponding to the major development areas of the LMFBR:

SYSTEMS ENGINEERING

- Mechanisms Under Sodium Test (MUST)
- Glove Box Environmental Test (GBET)
- Fuel-Handling Test Facility (FHTF)
- Sodium Vapor Deposition Rig (SVDR)
- Thermal Shock Test Models (TST-1)
- Sodium/Water Reaction (SWR-1)
- Hybrid Computer Laboratory

MATERIALS ENGINEERING

- Materials Test Loops (MTL)
- Mechanical Properties in Sodium (MPS)
- Brazement Test Loop (BTL)
- Pre-exposure Loop (PEL)
- Test Chamber Disassembly Dry Box Facility
- Vanadium Test Loops (VTL)
- Materials Structural Analysis Laboratory
- Welding and Brazing Facility

REACTOR ENGINEERING

- Thermo-fluids Laboratory
- Mechanical Laboratory
- Flow-Induced Vibration Test (FLINT)
- Control Rod Drive Mechanism Test (DMT)

FUELS ENGINEERING (Cheswick)

- Plutonium Laboratory
- Sodium Bond Development Facility

SODIUM TECHNOLOGY

- Fission Product Loops (FPL)
- Sodium Technology Laboratory

GENERAL PURPOSE FACILITIES

- General Purpose Loops (GPL)
- Steam/Water Loop (SWL)
- Sodium Cleanup Facility

TECHNICAL SERVICE LABORATORIES

- Supporting Analytical Laboratories
- Postirradiation Facility

SYSTEMS ENGINEERING

Heat Transport

Heat generated in the reactor core of the LMFBR is removed and transported by primary and secondary sodium loops to the shell side of a shell-and-tube steam generator. Because of the exothermic reaction and resultant hydrogen release that occur upon contact between water and sodium, the integrity and reliability of the steam generator become prime objectives in the development of the heat transport system. The Westinghouse facilities supporting this development area represent a strong base technology capability for the observation and testing of steam generator models to verify tube-to-tube-sheet configurations, transient thermal and fluid behavior, and sodium/water interactions.

The Steam/Water Loop (SWL-1) used in conjunction with the General Purpose Sodium Loop-1 (GPL-1) provides a facility for environmental testing of sodium steam generators. (Both facilities are described in a succeeding section.) The sodium loop furnishes the shell side flow up to 200 gpm, at temperatures up to 1200°F, and heat input up to 1 MW. The SWL-1 loop is capable of circulating 4000 lb/hr of 2700-psi feedwater to the tube side of the model at controlled temperatures up to 650°F. Each facility can simulate the thermal and flow transients that will be experienced in full-sized steam generators.

The immediate test program includes a series of Thermal Shock Test Models (TST) (Figure A-1) to evaluate tube-to-tube-sheet weldments and a three-tube, 1-MW Model to investigate thermal/hydraulic parameters.

One of the major considerations in the design and operation of sodium steam generators is the nature of sodium/water reactions. A basic understanding must be developed of the causes for tube and weldment failure and of the reaction phenomena accompanying such failure. The Sodium/Water Reaction Test Facility (SWR-1) (Figure A-2) permits study of the failure and propagation modes and reaction kinetics associated with sodium/water reactions in steam generators.

The SWR-1 is a simulated steam generator test chamber serviced by a small, low-pressure, liquid sodium loop that provides shell-side flow and a pressurized steam/water system for the tube-side environment. Tubes and weldments

Figure A-1a. Tube-Tubesheet Attachment-Thermal Shock Tests

Figure A-1b. Tube-Tubesheet Attachment - Thermal Shock Tests

Figure A-2. SWR-1 Sodium/Water Reaction Test Facility

with pre-programmed defects and flaws can be introduced into the test chamber and observed under conditions and geometries existing in a sodium-heated steam generator. The deterioration rates for the defective tube and subsequent propagation effects upon sound adjacent tubes can be measured. In addition, the facility can be used to explore burst type tube failures.

The SWR-1 will be operational late in 1969.

The steam generator designer requires transient test data and performance curves beyond the conditions normally expected in a sodium-cooled fast reactor power plant. Steam generator models will be tested under such extreme test conditions as to initiate failure. A third steam generator facility of smaller size is planned, especially designed to handle severe transients -- the Steam Generator Transient Facility (SGT-1). As well as providing sodium/water environmental conditions typical of the LMFBR, it will allow the imposition of severe temperature transients of $\pm 40^{\circ}\text{F}$ per second.

Fuel Handling

In addition to the usual constraints imposed by radiation and criticality, fuel-handling operations in the LMFBR are complicated by the physical and chemical nature of the sodium coolant: the opacity of the coolant requires that in-core operations be conducted without benefit of visual aids; a small, but measurable sodium volatility presents potential frosting and vapor deposition problems; the requirement to maintain the sodium fluid imposes a minimum temperature of $250\text{--}300^{\circ}\text{F}$ upon the operations; a high affinity of sodium for oxygen and moisture dictates the need for dry, inert gas atmospheres during exposure of coolant; and the excellent electrical conductivity of sodium can create unusual requirements for insulation and electrical devices.

Several types of fuel-handling systems have been advocated. Although no system has been fully proved in service, the hot-cell system has been selected for development because it appears to meet the most critical requirements of simplicity, plant availability, and maintainability. This system employs a large, shielded cell filled with an inert gas and located over the top of the reactor. To refuel the reactor with this system, the shield plug is removed, completely exposing the surface of the coolant, while the core remains

submerged in the sodium coolant. The fuel-handling machine extracts fuel from the reactor sodium pool and transfers it for radiation decay.

The major development tasks are to provide specialized electrical, mechanical, and pneumatic mechanisms with a high degree of reliability to operate in sodium and inert gas environments. The pursuit of the base technology informational needs is being conducted in three engineering rigs: Mechanisms Under Sodium Test Facility (MUST), Glove Box Environmental Test Facility (GBET); and Sodium Vapor Deposition Test Rig. The final phase of the engineering development will be the construction of a major-scale proof-test facility, the Fuel-Handling Test Facility (FHTF).

The Mechanisms Under Sodium Test Facility (MUST) (Figure A-3) provides a large sodium-filled test tank into which fuel-handling components, such as bearings, grippers, valves, and other electromechanical devices can be inserted for test at temperatures up to 1200°F. The sodium in the tank can be circulated through a purification system. Major components are a 30-inch-diameter stainless steel tank; a shell-and-tube heat exchanger to reduce the sodium temperature to effective cold-trapping temperatures; an 8-gpm electromagnetic pump; an electromagnetic flow meter, a dump tank, and air-actuated valves.

The Glove Box Environmental Test Facility (GBET) (Figure A-4) will be used to test mechanical and electrical components in an argon/sodium environment. Unlike the MUST facility, which can also be used for similar tests, GBET has the advantage of direct physical and visual access for easy insertion and removal of experiments. Typical tests conducted in GBET include determination of argon/sodium environmental effects on motors, switches, insulation, fabrics, paints, seals, and lubricants. The main component in GBET is the 48 x 40 x 29-inch glove box that houses a 12-inch-diameter stainless steel, sodium-filled tank. A sodium purification system, identical to that of MUST, is also provided.

The Sodium Vapor Deposition Test Rig is a smaller version of GBET with a glass bell jar substituted for the glove box. The facility is used to accumulate basic data such as correlations of sodium evaporation rates from sodium pools into argon environments.

Figure A-3. Mechanisms Under Sodium Test Facility

Figure A-4. Glove Box Environmental Test Facility

The Fuel-Handling Test Facility (FHTF) (Figure A-5) will demonstrate the feasibility of the LMFBR hot-cell-refueling concept and will provide the necessary full-scale operating experience with reactor components, refueling and equipment-handling machinery, environmental control systems, viewing equipment, etc., in a prototypical environment before final design and construction of the first LMFBR plant is completed. Further, after the first LMFBR plant is on-line, the Fuel-Handling Facility can be used for operator training and new component testing.

The FHTF is 40 feet long by 25 feet wide and has an over-all height of 100 feet. It will contain fuel-handling machinery, a reactor vessel mockup, sodium and argon service systems, an overhead bridge crane, manipulators, viewing windows, a closed-circuit television system, and systems for personnel access and equipment transfer. In addition, provisions are included for installing a shield-plug-handling machine, fuel canning and decanning equipment, additional manipulators, inspection equipment, periscopes, and a spent fuel pit at a later date.

Process Control and Instrumentation

Analytical models and computational methods must be developed to simulate the behavior of the LMFBR power plant and its components during normal steady-state and transient operation and during abnormal operation resulting from system or component malfunction. The stability, transient behavior, and interactions of systems and components as computed will provide the informational needs for the concepts selection, and design and development of the plant process control and instrumentation system. Obviously, the complexity of the analytical models and experimental simulations can only be handled by the most modern and flexible machine computational tools available.

The Hybrid Computer Laboratory (Figure A-6) consists of three major subsystems: two Applied Dynamics AD/4 analog computer consoles and display systems; a Scientific Data Systems Sigma-5 digital computer and peripheral equipment; and a dual high-speed interfacing system. While the AD/4's and the Sigma-5 represent extremely powerful computing systems in themselves, they have been

Figure A-5. Fuel Handling Test Facility

Figure A-6. Hybrid Computer Laboratory

interfaced in such a manner that greatly expanded flexibility, efficiency, and speed can be achieved for a wide variety of problems.

The AD/4 analog computers are of all solid-state construction and designed to generate solutions of high accuracy in both real and high-speed (compressed time-scale) modes of operation. Special emphasis has been placed upon dynamic accuracy of both computer and associated output and monitoring equipment. Flexibility exists in selection and expansion of computing capacity through plug-in modules. Full capacity of the two consoles is approximately 450 amplifiers and associated equipment.

The Sigma-5 is the first medium-sized machine to perform both real-time control and general-purpose computing, accomplishing both kinds of work at the same time with the same hardware and software concepts developed for the larger Sigma-7 computer. In addition to its speed, versatility, and full Sigma-7 compatibility, the computer provides the multi-use capability to handle simultaneously general-purpose computing in the background, multiple real-time operations in the foreground, and large numbers of concurrent input/output operations. The computer has a word-oriented memory (32 bits plus parity) for maximum efficiency, as well as data compatibility with IBM System/360 and modern communication codes such as EBCDIC or ASCII. The computer system is fully modular and can be expanded to a system approximately five times larger in high-speed core memory.

The Hybrid Interface System will provide monitoring and control linkage, logic signal linkage, and data linkage between analog consoles and the digital computer.

In operation, the analog system serves to perform the real-time portion of the simulation while the digital section provides the logical decision-making necessary for control of the program, as well as precision computation, function generation, and storage of information.

MATERIALS ENGINEERING

Corrosion and Mass Transfer Behavior

At the present time, there are insufficient data on mass transfer and corrosion rates of materials in sodium to design and fabricate many of the components

necessary for a dependable and economical sodium reactor system. In some respects, the study of materials behavior in sodium is simplified, as the primary mode of materials corrosion is simple dissolution of base elements and interstitials and mass transfer along temperature profiles.

It has been established in previous liquid-metal corrosion programs that the mass transfer rates of materials exposed to flowing sodium are functions of the sodium velocity, the temperature, the sodium impurity level, the heat flux, and the exposure time. Theoretically, the mass transfer rate should also be a function of the system temperature difference, the loop geometry, and the ratio of sodium volume to surface area.

To obtain valid corrosion and mass transfer data, it is necessary to closely approximate the environment encountered in a fast breeder reactor. The Materials Test Loops (MTL) (Figure A-7) have been designed to mock-up the LMFBR system by means of a design technique which equates the mass transfer driving force

$$\frac{\text{dissolution (or deposition) rate} \times \text{surface area}}{\text{Na volumetric flow rate}}$$

of the experimental rig to the plant system. In addition, the types, combinations and geometries of materials are duplicated.

The MTL loops are all identical in design and construction, and consist basically of hot and cold leg test sections, electromagnetic pump and flow meter, economizer, air-blast cooler, and cold trap circuit. The test specimens can be introduced or removed from the test legs during operation by mating an inert-gas dry box to flanges welded to valved expansion tanks located over the test sections. Thus, mass transfer data are obtained at periodic time intervals during a test run. Heat is introduced by external electrical heaters located over a portion of the hot leg section. The loops are made of 1/2-inch pipe and flow capacity is 5 gpm.

Upon completion of a test run, the entire loop is dismantled for metallographic examination. Currently, four MTL's have completed test runs of 2500 hours each and are in the second round of operations. Each loop simulates either the primary or secondary heat transport loop of the reactor, depending upon whether fuel cladding or steam generator materials are being tested.

Figure A-7a. Diagram of Materials Test Loop

Figure A-7b. Materials Test Loops

To examine the sodium corrosion resistance of several vanadium alloys and to pre-expose vanadium alloy specimens in sodium for subsequent testing of mechanical properties, two vanadium test loops (VTL-1 and VTL-2) are in operation. These loops, however, are less complex and less flexible operationally than are the MTL loops.

The design of the VTL loops provides for a somewhat different approach in sample removal. Each test loop will consist of three parallel test legs, sharing a common pump and cover gas system. An entire leg (containing specimens) can be removed from the loop after a specified exposure time, while the remaining legs continue to operate. After removal, the leg is cut into manageable-sized pieces and inserted in a dry box where the vanadium and stainless steel specimens are recovered.

MECHANICAL PROPERTIES IN SODIUM

Although there are currently considerable data being generated on the effects of sodium on the mechanical properties of LMFBR candidate materials, most of them are restricted to creep and tensile testing. Limited work has been done on the fatigue and stress relaxation characteristics of materials in actual sodium LMFBR environment. It is in this latter area that the Westinghouse mechanical properties test facilities are concentrated.

Two sodium Mechanical Property System Loops (MPS) (Figure A-8) and one Test Specimen Pre-exposure Loop (PEL) (Figure A-9) have been constructed and installed. They are similar in design except for test sections. The MPS test sections contain test machines that impose steady-state or cyclic stress upon test specimens submerged in flowing sodium. Continuous readouts of stress and strain data are also obtained. MPS-1 will obtain creep-stress rupture data for candidate materials while MPS-2 will gather stress relaxation data. Long-term exposure of test specimens for subsequent out-of-loop mechanical property testing will be conducted in the PEL loop.

Loop size and component description are similar or identical to the Materials Test Loops.

A vacuum purge dry box with a controlled atmosphere is utilized to disassemble sodium-contaminated environmental test chambers from the Mechanical Properties

Figure A-8. Typical Mechanical Property System Loop

Figure A-9a. Loop for Pre-Exposure of Mechanical Property Test Specimens

Figure A-9b. Loop for Pre-Exposure of Mechanical Property Test Specimens

Loops. The dry box will also be equipped with a welding torch to permit specialized welding under controlled environmental conditions. (Figure A-10)

Welding and Brazing Techniques

The development of brazing techniques and preparation of brazing samples for sodium service evaluation will be conducted in a high-temperature, high-vacuum brazing and heat-treating furnace. This furnace features molybdenum heaters, cold wall operation, and a capability of operating at 2800°F and 10^{-6} torr vacuum.

The brazing and weldment points will be evaluated in a small Brazing Test Loop (BTL) (Figure A-11), which is a slight modification of the Materials Test Loop design. It can operate at a maximum temperature of 1230°F, at flow velocities up to approximately 25 feet per second, and at a maximum pressure of 70 psig.

Materials Structural Analysis Laboratory

The Materials Structural Analysis Laboratory is a well equipped metallurgical examination facility. Equipment is available for metallographic sample preparation, mounting, polishing, examination and photography. In addition, test machines are also installed to determine mechanical properties such as micro-hardness. Darkroom facilities are planned for installation in 1969.

Water/Steam Corrosion Test Facilities

These facilities are being utilized to determine the susceptibility of Incoloy 800 and other potential steam generator materials to stress corrosion effects in high temperature aqueous environments containing chloride, caustic, and chemical control additives under conditions of static and cyclic stress and heat flux.

The facilities consist of a screening test facility and several autoclaves. Basically, the screening test facility, which performs high temperature burst tests on tubing, consists of a pressurizing system that stresses thin-walled tube specimens located in six separate test cells. Each test cell, containing a single specimen, is a 3-foot-long, 2-inch-diameter, schedule 40 nickel pipe, and is heated at the lower end and water cooled at the top to cause refluxing.

Figure A-10. Test Chamber Disassembly Dry-Box Facility

Figure A-11a. Brazement Test Loop

Figure A-11b. Brazement Test Loop

Three Inconel 600 autoclaves are being used to determine the limiting hydroxide concentrations for stress corrosion failure at 669°F and 2500 psi. Six specimens will be pressurized through six penetrations in the autoclave head. Three additional autoclaves are planned for chloride stress corrosion studies.

REACTOR ENGINEERING

Thermo-fluids Laboratory

The thermo-fluids laboratory is equipped to conduct thermal and hydrodynamic engineering experiments with both water and sodium on a bench scale. Typical experiments include: the evaluation of hydraulic models for parallel flow vibrations, fuel rod bowing, flow distribution, and mixing; the evaluation of thermal models for boiling and two-phase flow, and two- and three-dimensional heat transfer; and the study of steady-state and transient behavior of sodium bonding in fast reactor fuel rods.

The basic experimental test rigs in the laboratory are a small water loop and a small sodium loop. The water loop will provide a continuously variable flow of water or Freon 113 up to 160 gallons per minute. High accuracy instrumentation has been included for the measurement of pressure, temperature, and flow rate. Thermal tests may be conducted, as the loop has a heat input/rejection capability of 25 kW.

The sodium loop was designed for component, heat transfer and instrument tests, largely to complement and support more elaborate test setups in the General Purpose Loops 1 and 2. It also can be used for sodium boiling and two-phase heat transfer stability tests. The maximum design temperature and pressure are 1500°F and 50 psig, respectively. Provision is made for supplying 25 kW of electrical power to test sections. Heat rejection is accomplished by an air-to-sodium heat exchanger.

Hydraulic Test Facilities

In support of the PWR program, the Advanced Reactors Division operates a Hydraulic Test Facility (Figure A-12) to evaluate mechanical and dynamic effects in water of reactor-scale models. The facility, which consists of four loops

Figure A-12. Hydraulic Test Facility

(modeling a PWR reactor system), can deliver 8000 gpm at 100-psi pressure. Temperatures can be varied from 70 to 300°F.

Owing to some similarity of the hydraulic properties of water and sodium, many useful hydraulic static and dynamic tests on model LMFBR pressure vessel and internals can be conducted in this hydraulic facility. Higher temperature gradients in a liquid metal reactor require the use of three-dimensional flow distribution data in plena and fuel assemblies to accurately calculate temperature distribution in the coolant, structure, and fuel. A suitably sized FBR model complete with simulated fuel assemblies will be designed, fabricated, and tested in the Hydraulic Test Facility. The model size will range from 1/7 to 1/4 actual size, depending on the amount of design data desired. Water conditions will be selected which approach the kinematic viscosity equivalent to that of sodium at reactor temperature.

LMFBR fuel assemblies are subjected to a variety of severe environmental conditions. Flow rates as high as 35 ft/sec impose static and dynamic loads which have been evaluated experimentally. The flow Induced-Vibration Test Facility, FLINT, used in combination with the Hydraulic Test Facility, provides a water capability to investigate the vibrational characteristics of fuel element assemblies under these severe flow conditions. Basically, it consists of a fully instrumented test chamber installed in one of the water loops. A number of fuel rod clusters will be used to provide dynamic simulation of the flow channels surrounding the test rod cluster. The vibration spectrum can be obtained over a full flow range.

CONTROL ROD DRIVE MECHANISM TEST

As a part of the LMFBR development program, Westinghouse is currently testing and evaluating, in a liquid sodium environment, the adaptability of already developed PWR components.

Such a component, the PWR magnetic latch control rod drive mechanism, has been tested in an argon/sodium vapor cover gas environment. The Control Rod Drive Mechanism Test Facility (DMT-1) was used for this purpose (Figure A-13).

Figure A-13a. DMT-1 Sodium Loop

Figure A-13b. DMT-1 Sodium Loop

At the completion of over 1,250,000 operations of the mechanism, well above the estimated total number of operations expected in the life of a commercial reactor plant, investigation has shown the effects of sodium deposition on latch wear to be negligible.

Mechanical Laboratories

The Mechanical Laboratories provide bench-scale facilities for development and testing of small electromechanical devices, and contain specialized equipment for conducting photo-elastic stress simulation tests and for measurement of mechanical properties and vibration characteristics of rods, assemblies, and other components.

FUELS ENGINEERING

Plutonium Laboratory

The ARD Plutonium Laboratory (Figure A-14), located at Cheswick, Pennsylvania, has capabilities for producing mixed plutonium/uranium carbide fuel powder and pellets, and for fabricating and characterizing both carbide and oxide fuel elements. In addition to normal inspection capability, characterization capability includes a variety of novel metallographic techniques, particle-size measurement, and sodium-bond integrity techniques. Equipment for inspection and in-depth characterization of cladding is also available.

The purpose of the mixed carbide fuel (Pu,U)C development program is to develop base fuel technology on fuel performance, which includes both in-pile and out-of-pile testing. Nine pins of sodium-bonded (Pu,U)C fuel elements that have been fabricated at the ARD Plutonium Laboratory have been irradiated in thermal flux to burnup levels of 50,000 MWD/T. Sodium-bonded carbide pins for irradiation in fast flux are to be fabricated, and out-of-pile fuel/cladding compatibility tests are to be performed at the laboratory.

The Advanced Reactors Division is installing an oxide fuel glove box line at Cheswick with the capacity of providing up to ten pins per week for an oxide irradiation program. The mixed oxide programs include the design, design analysis, detailed specification, development, pilot-scale fabrication,

Figure A-14. ARD Plutonium Laboratory

characterization, and inspection of both 19-pin capsule and 37-pin, grid-type subassemblies for fast flux irradiation testing.

SODIUM TECHNOLOGY

Sodium Technology Laboratory

Several chemical unit operations may be considered as methods for the control and removal of impurities in sodium (atmospheric, corrosion products, and fission products). These operations include absorption, precipitation, filtration, chemical reaction, centrifuging, and distillation. The Sodium Technology Laboratory is equipped to conduct basic physical chemistry, thermodynamics, and kinetics experiments in investigation of these unit operations. In addition to the usual chemical apparatus, the laboratory is outfitted with inert-gas glove boxes and radiochemical hoods, which facilitate the handling of small quantities of sodium and radioactive tracers.

Fission Product Loops

Direct venting of fission gas and operating the fast breeder fuels to high burnups and at high specific powers may introduce fission products into the sodium coolant. Also, mass transfer introduces another type of contaminant into the coolant in the form of neutron-activated corrosion products. To study and evaluate the effectiveness of various methods for removing these contaminants, two sodium loops are being operated, Fission Product Loops 1 and 2 (FPL-1, -2).

FPL-1 is a 1/2-inch-diameter, 2-gpm main loop with a 1/4-inch-diameter, 0.1-gpm cold-trapped bypass loop. Typical tests conducted in this loop investigated the mechanisms involved in the apparent removal of cesium by cold trapping (Figure A-15).

FPL-2 (Figure A-16) is a more versatile test rig, as it includes a primary loop, two cold-trapping circuits, and a hot-trapping purification circuit. The primary loop, fabricated of 1/2-inch-O.D. stainless steel tubing, consists of a 2-gpm electromagnetic pump, 20-kW immersion heater, electromagnetic flow meter, cooler, valves, sump tank, and expansion tank. The cold and hot trap loops are identical except for the internals of the traps. Each purification

Figure A-15. Fission Product Loop

Figure A-16. FPL-2 Arrangement

loop consists of the trap, economizer, electromagnetic flowmeter, and valving. The cold traps are vertically mounted, 4-inch-O.D. tanks with internal baffles. Cooling is accomplished by forced air circulation through a cylindrical sleeve installed around the trap. The hot trap is a 2-inch-O.D. tank, 21 inches in overall length, packed with 0.004-inch-thick zirconium foil. An electric clamshell heater provides heat input to the trap.

GENERAL PURPOSE FACILITIES

General Purpose Loop-1

The general purpose loop facilities are large sodium-pumped loops that provide broad capabilities for conducting tests ranging from basic studies in heat transfer, hydraulics, and sodium technology to in-sodium testing of LMFBR components.

General Purpose Loop-1 (GPL-1) (Figure A-17) is a dynamic, high-temperature system that provides sodium to test sections at temperatures up to 1200°F and flows to 200 gpm. The test sections are installed either in a test cubicle located outside the main building or in a test enclosure located adjacent to the loop. The exterior test cubicle is a silo-type structure that accommodates large test components such as steam generator models. The test enclosure is primarily employed for the basic studies of heat transfer and hydraulic phenomena.

The GPL-1 system is fabricated of Type 304 stainless steel seamless pipe. The main flow path is through 2-inch pipe, while auxiliary systems are 1/2 and 1-inch pipe. Flow is obtained from a 200-gpm linear induction electromagnetic pump. To provide system flexibility, an air-to-sodium finned-tube heat exchanger is utilized as either a 1-MW gas-fired heater or forced-air cooler, depending upon test section requirements. Purification and sodium sampling facilities are included as a part of the basic GPL-1 system.

The Steam/Water Loop

The Steam/Water Loop (SWL-1) (Figure A-18) complements the General Purpose Loop-1 to provide a combined facility for environmental testing of sodium-heated steam generators. The loop provides feed water to steam generator models

Figure A-17. General Purpose Loop (GPL-1)

Figure A-18. Steam/Water Loop

at controlled pressures and temperatures, and condenses the superheated steam received from the models. The loop is able to simulate the transients that will be experienced in full-sized generators.

A centrifugal pump circulates the high-pressure feed water through purification beds to the secondary side of a tube-in-tube regenerative heat exchanger. The water then passes through a gas-fired preheater where the feed water can be adjusted to a maximum of 650°F. It is then piped to the tube side of the sodium-heated steam generator model located in the test cubicle. The superheated steam from the models is condensed in the finned tubing of an air-cooled heat exchanger, and the condensate is passed through the regenerative heat exchanger and an aftercooler. The loop incorporates a let-down and pressurized make-up system, plus temperature, pressure, and flow measurements around the circuit.

General Purpose Loop-2

General Purpose Loop-2 (GPL-2) (Figure A-19 and A-20) is the largest sodium engineering loop at 2000-gpm flow rating. It is sized to accommodate the thermal and hydraulic testing of full-scale LMFBR fuel assemblies, control rod mechanisms, and other large sodium components. The test sections are installed in the Building C penthouse which features an overhead clearance from crane hook to floor of 65 feet. In addition, GPL-2 can simultaneously test smaller test sections requiring flows up to 2000 gpm.

The main test loop of GPL-2 features 6-inch stainless steel piping with flow provided by a 2000-gpm linear-induction electromagnetic pump. An air-to-sodium heat exchanger in the main loop serves as a dual-purpose heater/cooler. With direct gas-firing of the incoming air, the heat exchanger can input up to 1.25 MW. During the cooling mode of operation, the same heat exchanger can remove 2.5 MW.

Two auxiliary bypass loops across the pump furnish the functions of coolant purification and sodium sampling and analysis. The purification auxiliary loop, the Sodium Chemistry Control System (SCCS), consists of economizer, cold trap, plugging meter, and hot trap. The Sodium Sampling System (SSS) will contain on-line analytical instruments and sampling apparatus for the removal of sealed sodium sample tubes.

Figure A-19. General Purpose Loop (GPL-2)

Figure A-20. General Purpose Loop (GPL-2)

Sodium Clean-up, Disposal, and Storage Facility

As a consequence of the experimental nature of the sodium facilities, experimental loops and test sections frequently must be dismantled and the sodium removed and discarded. A Sodium Clean-up, Disposal, and Storage Facility will be designed and constructed in 1969 to facilitate these operations.

The facility will provide floor space for the safe disassembly of experimental rigs. Apparatus will be available for steam cleaning and alcohol dissolution of adherent sodium from pipe and mechanical components. Dry, ventilated areas will be provided for the storage of sodium.

Small quantities of sodium can be easily disposed of by reaction with steam or alcohol-type organics. Larger quantities of sodium can be safely disposed of by means of controlled burning. A gas incinerator, complete with fume scrubbing apparatus, will be designed for the facility.

TECHNICAL SERVICES

Technical Service Laboratories

The development of nuclear power and its growth as an energy source for producing electricity brought about a need for unusual and special analytical work. The ARD Technical Service Laboratories have been designed and equipped to meet this vital need for comprehensive service analysis in the nuclear development field.

The Analytical Chemistry Laboratory can perform a variety of analyses on metal alloys such as stainless steels, Zircaloy, Inconel, vanadium, etc., uranium oxides and carbide fuels, aqueous solutions from corrosion testing systems, and inorganic materials of unknown composition.

The Gas Analysis Laboratory (Figure A-21) has the capability of analyzing gas samples for all gaseous impurities, as well as the liberation and subsequent analysis of gases from nuclear fuels, metals, ceramics, and all the other materials involved in nuclear reactor technology. The performance of other experiments involving gas analyses of a qualitative or semi-quantitative nature are also possible in this laboratory.

Figure A-21. Gas Analysis Laboratory

The Radiochemistry Laboratory provides radiochemical services that include general radiochemical analysis, provision for postirradiation analysis of neutron flux dosimeters, provision of special forms of radioactive tracers, design and field use of systems for the gamma scanning of nuclear fuel, and design of experiments involving radioactive tracers.

The Spectrographic Analysis Laboratory determines the composition of materials (primarily inorganic) by causing them to emit or absorb electromagnetic radiation characteristic of the element within the material. The wave length of this radiation observed by the instruments in the laboratory ranges from the near infra-red region, through the visible and ultraviolet regions, and down to the high energy X-ray or soft gamma region. The concentration range of the elements determined varies from greater than 50 percent to less than 10 parts per billion with the size of the sample subjected to excitation varying from several grams to a few micrograms.

The Liquid Metals Analytical Laboratory performs the analysis of interstitial impurities in various liquid metals, such as sodium, potassium, and NaK. The laboratory has the ability to determine oxygen, nitrogen, and carbon on a routine basis.

The Mass Spectrometry Laboratory has the capability of analyzing samples by means of thermal or spark-source ionization. The detection sensitivity of this technique is high, whether defined in terms of the low concentration of the impurity detectable in a matrix or in terms of the total amount of sample needed to detect the impurity.

Postirradiation Facility

The Postirradiation Facility (Figure A-22) is capable of performing complete nondestructive and destructive postirradiation examination of fuel rods, experimental capsules, and associated nuclear reactor components.

The facility currently consists of three high-level-radiation hot cells, one metallographic cell, a high-bay Transfer Building equipped with a bridge crane and containing a water pit for spent fuel storage, and a mechanical testing facility.

Figure A-22. Postirradiation Facility

RELATED FACILITIES OF OTHER WESTINGHOUSE DIVISIONS

Nuclear Energy Systems

Nuclear Energy Systems, one of the three major functional groups of the Westinghouse Power Systems organization, (Figure A-23) is responsible for the over-all Corporate participation in the commercial nuclear power industry. Each of the operating divisions has specialized responsibilities that contribute to the leadership position that Westinghouse holds in the atomic power industry.

Figure A-23

From the expertise of these Nuclear Energy Systems Divisions, the Advanced Reactors Division will receive support in the LMFBR Development Program.

PWR Systems Division

The Pressurized Water Reactor Systems Division, which is responsible for the development, design, marketing, and procurement of pressurized water reactor nuclear steam supply systems, has vast experience in reactor design and construction. From this experience, the PWR Division will provide consultation to the Advanced Reactors Division in all aspects of plant design and development.

Nuclear Fuel Division

The Nuclear Fuel Division (NFD) manufactures nuclear fuel and provides related fuel services for commercial power reactors. From this division, supporting consultation will be available to ARD in fuel design, fabrication and reprocessing.

The NFD Plutonium Fuels Development Laboratory at Cheswick, Pa. is a pilot facility to develop the production processes necessary for the economical fabrication of both thermal and fast reactor plutonium fuels. The laboratory will develop manufacturing methods for plutonium recycle fuel for light water reactors and provide the facilities to fabricate specialized FBR prototype fuel rods for irradiation and testing in the LMFBR development program. The plant will have sufficient capacity to manufacture the first LMFBR core and replacement zones for the Westinghouse Demonstration Plant.

The Atomic Equipment Divisions

The Electromechanical Division is responsible for the development, design, manufacturing, and marketing of specialized hydraulic, mechanical, and electrical apparatus for nuclear reactors. The specialty products related to the LMFBR development program include pumps, valves, control rod drive mechanisms, flow meters, and electro-mechanical components for use in pumped liquid metal systems as well as reactor internal structural parts that would be subject to a liquid metal environment.

The Specialty Metals Division at Blairsville, Pa., is responsible for the manufacturing of the zirconium tubing that is used to contain the UO_2 fuel

in water reactors. In addition to zirconium tubing, the Specialty Metals Division manufactures Inconel tubing for steam generators and other heat transfer apparatus, magnetic alloys, glass and ceramic-to-metal sealing alloys, and high-temperature alloys.

Large Components Divisions

The Tampa Division is responsible for the design, manufacture, and marketing of steam generators, pressurizers, and other heat transfer apparatus for nuclear power plants. Now a part of the Nuclear Energy Systems, the Tampa Division, originally an expansion of the Heat Transfer Division, has shared the extensive nuclear power experience of the former Westinghouse Steam Divisions. Plant facilities at Tampa provide for low and high-temperature mechanics, welding research and development, and chemical analysis.

Since steam generators are one of the major development areas in the fast breeder reactor program, close coordination of the development, design, and testing of steam generators is maintained between the Tampa Division and the Advanced Reactors Division.

The Pensacola Division is responsible for the design and manufacture of internal structural parts for nuclear reactors. The reactor internals, fabricated of stainless steel, are used to position and support the nuclear fuel assemblies and as guideways for reactor control rods. The precision machining, fabrication, welding and assembling that are required in the manufacturing of these light water reactor internals will also be necessary for the reactors of the LMFBR plants of the 1970's.

WESTINGHOUSE TELE-COMPUTER CENTER

Today's ever increasing demands on engineering computation and simulation in the design and analysis of nuclear power plants require greater computer capacity and versatility than ever before. To meet these computation needs, the ARD has direct access, by a remote computer terminal, to the extensive digital computer facilities of the Westinghouse Tele-Computer Center.

Connected to a CDC-6416 input/output device of the Tele-Computer Center via wide band circuits, the high-speed terminal at Waltz Mill is capable of driving a remote high-speed printer, card reader, and card punch at rated speed. This capability permits the remote user at Waltz Mill the same service, turnaround, and operational procedure as a local user of the Tele-Computer Center. The CDC-6416 input-output device is connected via separate high-speed channel-to-channel connections to two CDC-6600 digital computers. After arranging incoming work for the terminal, the CDC-6416 passes the work to the appropriate CDC-6600 main processor for execution, arranges the computed results, and transmits them back to the terminal output.

By having this terminal at Waltz Mill, ARD is able to perform engineering and scientific calculations on two of the fastest commercially available computers in the world.

CENTRAL RESEARCH LABORATORIES

As the central research and development activity of Westinghouse, the Central Research Laboratories have two primary objectives:

1. Expedite development of new scientific knowledge and its translation into new products, processes, materials, and systems.
2. Provide the highly diversified technology required for the wide range of advanced components and systems produced by Westinghouse.

In meeting these objectives, the Central Research Laboratories give invaluable service to the Nuclear Energy Systems Divisions, aiding extensively in the pioneering of new developments in atomic power technology.

In the Central Research Sodium Laboratory, the research focuses upon basic studies in the chemistry of sodium. These studies have included those of sodium impurity systems (thermodynamic, solubility, reaction rates, and equilibria) and the development of an on-line impurities meter for fast breeder reactors. Such studies of sodium are important in the development of the primary coolant system for the LMFBFR.

Other important Central Research development programs underway in support of the ARD include component life-time in a sodium vapor hot cell, in-core instrumentation for liquid metal reactors, under sodium scanner, fission gas effects on cladding and fuel temperature, fuel and cladding interaction, and evaluation and stress analysis of reactor vessel and head. All of these programs are vital to the timely development of the LMFBR.