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FAST FLUX
TEST FACILITY
PROJECT REPORT

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FOREIGN TRIP REPORT

L. M. Finch

DECEMBER 1966

FAST FLUX
TEST FACILITY
PROJECT REPORT

AEC RESEARCH &
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December, 1966

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PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

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FOREIGN TRIP REPORT

L. M. Finch

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FOREIGN TRIP REPORTDecember 1-21, 1966INTRODUCTION

During December, 1966, foreign sites involved in fast reactor technology were visited to obtain information expected to be useful in the conceptual design of the Fast Flux Test Facility (FFTF). The sites visited were at Thurso, Scotland, the site of Dounreay Fast Reactor (DFR) and also the site of Prototype Fast Reactor (PFR) that is presently under construction; Risley, England, (near Manchester) where the principal engineering and development work for the British prototype fast reactor (PFR) is being performed; Mol-Donk, Belgium, where the BR-2 thermal reactor is in operation with a skewed core geometry; Karlsruhe, Germany, where the conceptual design work for a German sodium cooled fast reactor is being performed; Saclay, France, where design for the French fast reactor (Phenix) is being performed; and Cadarache, France, the site of the Rapsodie fast breeder reactor. At each of the sites, inquiry was focused upon the reasons for concept selection and upon understanding the application of the selected concepts to fast reactors. Typically, the reasons for concept selection for a particular reactor were not well documented and consequently, the reasons were quite difficult to recall except for more obvious conceptual choices. For instance, the choice of many (24) small heat transfer circuits for the Dounreay Fast Reactor can be established as state-of-the art for pumps and heat exchangers in Britain at the time of the DFR construction. The factors which contributed to the choice of down-flow coolant in the DFR core are more difficult to recall.

PURPOSE OF TRAVEL

Safe, highly reliable and predictable plant performance for the Fast Flux Test Facility (FFTF) is required to develop and demonstrate integrity of fuels and materials for the Liquid Metal Fast Breeder Reactor (LMFBR) program in a timely manner.⁽¹⁾ Development of reactor design concepts exhibiting a range of capabilities and costs has been requested of Pacific Northwest Laboratory (operated by Battelle Northwest) by the AEC.⁽²⁾ Reactor designs which meet the irradiation testing requirements within a feasible concept are limited in number and the concepts themselves are difficult to execute. While an extensive FTR design development program is in progress in the United States, it is important that potential benefits or hazards to the FFTF program be identified in European technology as early as possible. Essential decisions will be made concerning reactor and irradiation testing concepts shortly after the beginning of the calendar year. The decisions must include consideration of the extensive European experience in both fast reactors and fast flux testing prior to firm commitment to the design and construction of a particular preferred FFTF conceptual arrangement.

The Pacific Northwest Laboratory has been assigned FFTF program direction.⁽²⁾ A major area requiring assurance of complete performance, reliability, and safety is the reactor vessel system. The trip was intended to yield design and Construction information concerning the British Dounreay Fast Reactor (DFR), the British Prototype Fast Reactor (PFR - under construction), the German Karlsruhe Fast

(1) Shaw, M., "Fast Breeder Reactor Program in the United States," presented at the British Nuclear Energy Society Conference on Fast Breeder Reactors, London, England, May 17, 1966.

(2) Shaw, M., "Fast Flux Test Facility - Program Direction," Memorandum to Donald G. Williams, January 28, 1966.

Reactor, and the French Rapsodie Fast Reactor. Since each of these reactors has some associated experimental irradiation test function, the performance evaluation of such tests is of critical interest to the FFTF program. The Belgian BR-2 Materials Test Reactor, while not a fast reactor, has a unique core oriented system. A simplified skewed array is being considered as a candidate reactor core concept for the FFTF because of improved access to irradiation test facilities and reactor control mechanisms and to facilitate instrumentation of core subassemblies.

DOUNREAY, SCOTLAND - December 5 and 6, 1966

Persons visited:

T. A. Parry, Deputy Director, Dounreay Fast Reactor

J. Gray, DFR Operations

J. G. Walford, DFR Operations

Dr. J. S. Bradley, Head of Chemistry Division

L. R. Williams, Head of the Metallurgy Division

The morning of December 5 was consumed by a resume' of the DFR program as described by Mr. Parry, Deputy Director, followed by an extensive discussion with Mr. Gray who described the DFR operating experience. During the afternoon of December 5, Mr. Walford conducted a tour of the four-inch test loop facility located north of the DFR. He also provided a tour through the Materials Test Reactor which is on the same site with the DFR. We did not visit the PFR site because of the press of time and because the present progress of construction work is limited to excavations for the reactor cavity.

The morning of the sixth was consumed by more discussions of the DFR and

some applications of DFR experiences to PFR. Because of a midafternoon flight time to Manchester, the afternoon activities of December 6 were limited to a rather quick visit to the hot lab examination facilities supporting the DFR, and a short summary discussion with the deputy director. An organizational chart for the Dounreay Experimental Reactor Establishment is provided as Appendix II to this report.

Of special interest at the Dounreay site was an organizational component called the Irradiation Project Office (not directly associated with the DFR operations). The Irradiation Project Office was charged with the responsibility for (1) managing all aspects of the experiment design, fabrication and testing, (2) correlation of all test information, and (3) preparation of reports and dissemination of the test information. The Project Office functions as the responsible contact with sponsors desiring experiments in the DFR. During a three-year period the Project Office has handled 400 tests in the DFR. The interjection of a special irradiation project organization between the sponsors of the experiments and the reactor operations group appears to have merit.

Representatives of the Irradiation Project organization are certain that the sponsors of experiments should not be permitted to design and install the experiments in the reactor independently. The success of an experimental program depends to a large degree upon the accumulated experience and know-how of the Project Office.

Observations and Notes

- (1) PFR test specimens irradiated in the DFR have achieved burnups to 8% of the heavy atoms.
- (2) The DFR has experienced no leaks or other failures in the double-wall pipe installations in the primary coolant system.

Even though the DFR coolant systems have been successful, the British would not use the doubly contained piped concept for sodium cooled reactors again. Lacking failures in double walled piped systems, the British are compelled to assign high reliability to the double walled concept despite fabrication and inspection difficulties. They are not, however, assured that the size of the data sample is sufficient for reliability predictions.

- (3) The steam generator for Dounreay has failed on the NaK tube side at least once. The prevailing practice is to remove the affected section of the steam generator and derate the equipment accordingly.

The unique design of the Dounreay steam generator has interesting potential application to the FFTF or other systems which have a large temperature drop between the heat source and the cool sink. Essentially, the steam generator element design consists of four parallel pipes arranged in a square pitch and thermally coupled by thin copper plates pierced with four holes and tightly compressed along the full length of the heat transfer pipes. Thus, the steam generator element becomes a laminated copper block pierced with four pipes, two of which carry the NaK coolant from the reactor and two of which carry the water as liquid or vapor. The purpose of the laminated system is to allow venting of either the water or NaK in the event of a pipe failure without causing failure of the other pipes in the generator section. The failed DFR steam generator did demonstrate that the laminated design prevented progressive failure of steam generator sections. The DFR operations personnel complained about the relative thermodynamic inefficiency of the steam generator design caused by the required large

temperature drop through the copper plates. The inherent inefficiency of the FFTF as a waste heat machine might permit the use of a steam generator or a water heater design similar to that used on the Dounreay steam generator. If the air heat dump of the FFTF is abandoned for any reason, the Dounreay steam generator design might be a useful alternative.

- (4) The DFR has coolant mixing problems at the inlet (top of core) as evidenced by identification of sodium streams of dissimilar temperatures after passage downward through the core (even at full power). It is difficult to maintain all of the 24 primary coolant circuits at the same inlet and outlet temperatures for various reasons (such as the derated steam generator in item 3 above). Any temperature difference among the 24 primary circuits is reflected at both inlet and outlet of the core in the vicinity of the primary coolant circuit connections to the reactor vessel.

It was not indicated by the British that temperature streaming as described had ever caused significant distortion of the reactor vessel. On the other hand, they could provide no assurance that the reactor vessel in the vicinity of the 24 primary circuit inlet nozzles had not suffered severe thermal cycling because of excessively cool or excessively hot coolant streams entering the reactor vessel.

- (5) The DFR EM primary pumps require pressurization of the coolant (NaK) to control electromagnetically induced vibrations on the pump channel walls (about 0.08 inches thick).
- (6) The original fixed thermocouple installations at the core and blanket subassembly discharge regions (at the bottom of the core) do not give

representative outlet coolant temperatures for the individual sub-assemblies. Instead, the thermocouples indicate mixed mean temperatures of nearby subassemblies. DFR operations, however, finds the subassembly thermocouples to be desirable and maintains continuous monitoring of the temperature readings.

- (7) The swelling of blanket subassemblies at about 100 MWD/ T was a surprise to DFR operations. The swelling is attributed to metallurgical growth of the blanket fuel as a result of thermal cycling across the phase transformation temperature. Cause of the temperature cycling has been identified in the flow reversal of NaK coolant when the coolant frictional pressure loss in a blanket subassembly is less than the natural convection head developed in the subassembly flow channel during reactor operation. Since the specific heat generation in the blanket subassemblies is small, the design flow rates and pressure losses are also small, leading to potential flow reversal.
- (8) The sodium test facilities at Dounreay are developing experience with sodium leaks and their consequences, calcium removal from sodium coolant, and techniques for detection of nucleate boiling in sodium.
- (9) DFR operations indicate the experimental charge rates in the DFR are 200 L per day per channel in the core zone and 80 L per day per channel in the blanket zone. Present operational cycle is 55 days continuous operational period followed by a 25-day shutdown period for refueling. This operational cycle contrasts to the previous cycle which was 90 days operational period and 35 days shutdown period. The DFR is operating on about a 70% plant factor.

- (10) Even though the DFR was not originally designed as a test reactor and has been able to accept tests only by core modification, the volume of test data and experimental output from the DFR is astounding. The British have developed sophisticated techniques for performing in-core experiments including measurement. Capsule tests predominate in the DFR experiments and techniques for interim examination (discharge, examination, and refueling of a specimen during a single shutdown period) are fully developed. The extensive hot cell examination facilities at Dounreay are barely able to keep up with the experimental output of irradiated specimens from the reactor.
- (11) The PFR fuel and material tests in the DFR are essentially completed. At the time of the visit, the British were soliciting sponsors for tests in the Dounreay reactor.

RISLEY, ENGLAND - December 7, 8 and 9, 1966

Persons visited:

Dr. Nettley - RML, Culcheth
Dr. Mechele, RML, Culcheth
Mr. P. B. F. Evans, REL
Mr. D. Taylor, Primary Circuit
Mr. P. L. Riley, Reactor Design
Mr. Lothwaite, Primary Circuit
Mr. G. Jackson, Core
Mr. Webb, Fuel Handling

The morning of December 7 was spent in discussions with Mr. Evans of the Reactor Engineering Laboratory. Just before noon on December 7, a visit was made to Culcheth a few miles from Risley where the materials research and development

laboratories are located. During the afternoon of December 7, initial discussions were held with the reactor design personnel. Mr. Riley led the discussions which consisted of a series of meetings extending over the next two days dealing with specific systems in the PFR. The theme of all meetings with design personnel was the assured status of the PFR design. Both the laboratory and design groups at Risley were emphatic in their support of the closely integrated development and design organizational structure at Risley.

Observations and Notes

- (1) The British have selected a size for the PFR that is definitely prototypical of larger (1000 MWe or greater) liquid metal fast breeder reactors. They expect the PFR to be a successful demonstration unit, i.e., an economical power producer, while also functioning as a prototype LMFBR. It was obvious from discussions that a continuing conflict between prototypic features and more economical demonstration features had been encountered throughout the PFR design. It was also obvious that no established policy of preference for prototypic or demonstration features had prevailed.

- (2) Reactor Vessel

The British have settled upon a "tank" type reactor concept which is similar to the concept strongly advocated by GE in the U.S. The "tank" concept is so-called because the entire primary heat transfer system (including pumps, intermediate heat exchangers, valves, and minimal piping) is submerged along with the reactor in a sodium pool formed by a large vessel or "tank". The dimensions of the PFR vessel are typical; 38 feet diameter by 50 feet height.

This choice of reactor concept was reached by a tedious evolutionary procedure of concept evaluation. The starting point for the study sequence was a piped reactor vessel and primary circuit arrangement using optionally either three or four main primary circuit branches. This initial piped concept fitted well with the British experience at Dounreay (except, of course, the PFR heat transfer circuits were less numerous and larger than the Dounreay primary apparatus). During evaluation of the piped circuit, a difficult condition was recognized; namely, that the economic and engineering considerations of piped systems became increasingly uncertain as the reactor heat rating increased. While feasible and reasonable piped system designs could be developed for reactors the size of the PFR, reactors rated at 1000 MWe and larger were much less suited to the piped arrangement. In general, while the reactor vessel does not increase significantly in diameter as the core power rating is increased, the size and number of coolant pipes increases until there is insufficient surface on the vessel wall at the coolant discharge zone to accommodate the discharge nozzles. In addition, the large pipes have such great bending stiffness that long tangent lengths and heavy reactions at anchors must be provided to compensate for thermal expansion.

Thus, while the piped system was acceptable for the PFR at its 250 MWe rating, it did not seem prudent to the British to embark upon a prototype concept which held little promise for the liquid metal cooled fast breeder reactors of larger and more economical size.

The next step in the PFR concept evolution was a close coupled piped system utilizing large piping bellows instead of piping flexibility

for thermal expansion compensation. The negating problems for the bellows coupled concept were (1) poor experience with thin walled bellows in liquid metal environment, (2) poor potential for in-service repair or replacement, and (3) excessive development effort required.

The third evolutionary concept was the "satellite tank" concept wherein the pumps and heat exchangers were installed in three tanks removed from the reactor vessel and connected by large pipes (concentrically nested for coolant supply and discharge) to the reactor vessel. Alternate methods of compensating for thermal expansion were bellows and a roller or rod suspension of the satellite tanks which permitted the tanks to move radially with respect to the reactor vessel. The advantages of the satellite tank concept did not offset the previously listed deficiencies of bellows. The moving tank alternate was rejected because of anticipated difficulties with the movement of large pump and heat exchanger upper parts within a stationary reactor or cell shield plug.

The final choice for the PFR concept is a large single tank into which the reactor, pumps, and intermediate heat exchangers are suspended. The tank itself is devoid of all piping connections below the point of support at the cover.

The British recognized that the tank concept exacts a penalty in that it represents a commitment to a unified "whole" system which must be designed and analyzed as a single problem. Where the piped system is somewhat disconnected, at least to the extent the pipes are flexible, the tank system components cannot be treated separately in time or in function. Since the tank concept choice was made at an early date

in the PFR design, there has been time for the British to perform the analyses and experiments needed to arrive at the "unified" solution. Specifically, they claim to have solved (1) the DBA blast containment within the tank, (2) the tank support problem utilizing only the tank shell, and (3) the sodium flow problems in the tank under all operating modes.

(3) Core Structure Arrangement

The general arrangement of the fuel rod subassemblies in the PFR might be described as a modified "daisy" or free cantilever system. Groups of six fuel subassemblies are forced into interference fit against a central control rod subassembly. Thus, each core bundle of seven subassemblies including a central control subassembly tends to function as an integral unit. Detailed calculations for the response of such composites to changes in reactor temperature or reactor power were not available. In the ideal "daisy" arrangement the fuel subassemblies are free to bow outward as a result of the axial temperature gradient. Since the bowing or outward deflection of the fuel subassemblies increases with increasing power, the "daisy" arrangement provides a desirable negative power coefficient of reactivity. In discussions with the British, it was not clear how the anomalous response of the long, unfueled control rods could be predicted at the central position of a seven subassembly cluster.

A major concern in the tank type reactor system is the potential for activation of the secondary coolant at the intermediate heat exchangers by neutron irradiation from the core. Neutron flux is not sufficiently attenuated by sodium in the distance between the core and

the secondary coolant in the IHX to prevent activation; therefore, special neutron shielding must be provided. In the PFR, the British chose to include shielding in the upper ends of the radial neutron shield in the region of the primary coolant discharge to the heat exchangers. This choice was made in lieu of shielding extensions upon the fuel subassemblies. The reasons cited for not including shielding on the fuel subassemblies were as follows:

1. Increased coolant pressure drop for fuel subassembly with shielding
2. Increased length of fuel subassembly to include shielding
3. Increased reactor height to accommodate rotating plug refueling of increased length subassembly.

(4) Reactor and Core Materials

The materials group at Culcheth was more confident in its technical position, if possible, than the other design groups. The 316 stainless steel has been chosen as the PFR structural material. The Culcheth group had no reservations about the use of 316 stainless steel up to the design temperatures of the PFR. Further, they showed little or no interest in the stainless steel-sodium interactions observed by Fred Smith of ANL. Smith had discussed his findings with the Culcheth group during the week previous to my visit. The general attitude was that Smith's observations were anomalous and probably resulted from unique exposure conditions not reported by Smith and not typical of reactor environment. To support the contention, Dr. Nettley showed photographs of fuel rod sections which showed no intergranular penetration of sodium even in regions of a sodium inert gas interface. Further,

the research group emphasized that in their extensive studies of stainless steel in sodium they had no direct observation of the phenomenon reported by Smith.

(5) Intermediate Heat Exchangers

The PFR employs three primary circuit loops, each consisting of two intermediate heat exchangers and one pump. Since the British are definitely building a prototype reactor for the large LMFBR and not just a power demonstration unit, the use of two relatively small intermediate heat exchangers seemed anomalous in an otherwise prototypic design. The reasons presented by the British were:

- a. Large sodium heat exchangers are not in their experience, hence, caution is dictated;
- b. Larger IHX's would have increased the diameter of the tank; and
- c. Although the 1000 MWe and LMFBR will require larger IHX's than the PFR, it is likely they will be distributed in pairs as in the PFR (hence, the IHX's are prototypic in arrangement, if not in size).

The PFR IHX will be a shell and tube heat exchanger arranged for vertical downflow of primary coolant on the shell side and upflow of secondary sodium on the tube side. The British were confident their heat transfer calculations were sufficiently correlated by experiment to assure the IHX design. The relatively poor heat transfer coefficient predictions observed elsewhere for low velocity, shell side sodium flow did not concern the PFR designers noticeably. The available pressure to produce coolant velocity on the primary side is sharply limited by sodium coolant levels while no similar restrictions exist for the

secondary circuit. (See Appendix IV for sodium heat transfer data acquired at Risley).

During the discussions relating to IHX arrangement and support, an interesting condition was exposed. In plan view, the six IHX's are arranged in three groups of two closely spaced units which are connected to the core shroud by a flow baffle resembling a "three-leaf clover". The pumps are freely suspended in the three spaces between the "leaves". The flow baffle, which separates the hot core discharge coolant from the main contents of the tank, is subject to low mechanical loads and to small hydrostatic pressure differentials, hence, it is a relatively light, thin-walled assembly. During analysis, it was determined that the vertical skirt of the baffle would undergo a buckling deflection under severe temperature transients (such as a scram). The buckling was a waveform with an amplitude of about two inches. Since the skirt is not restrained, the buckled condition did not incur large stresses and consequent material damage. Since "tin-canning" with two inch deflection was not esthetically pleasing, stiffeners were added to the skirt design to maintain its shape during the temperature transients. However, esthetics were abandoned when excessive stresses were calculated for the restrained skirt. The present design is unstiffened.

(6) Primary Coolant Pump

The British intend to issue requests for "tenders" (bids) for primary pump procurement during December. The primary pump is a vertical sump centrifugal type with a free surface barrel and nominal rating of 15,000 gpm at 100 psi. The pump is quite similar to Byron Jackson apparatus in arrangement. The unit is equipped with pony motor and

inertial fly-wheel to provide shutdown coolant circulation and loss-of-power flow during scram events. The British seem to rely upon their commercial pump manufacturers for pump development much as we do in the United States. For instance, they plan to use a hydrostatic, sodium-lubricated journal bearing on the lower position on the pump shaft (just above the rotor) in lieu of a hydro-dynamic bearing. When pressed for an explanation of the choice, it became apparent that the British were relying upon the commercial manufacturers' recommendations. The vertical orientation of the bearing and the journal and the lack of lateral restraint were cited as the reasons why hydrostatic bearings could be applied with confidence in this application. British tests indicate that no pump starting problems should develop with the hydrostatic bearing. No experimental or analytical data was presented to cover the case of bearing-journal interference resulting from thermal distortions of either the shaft or the pump casing.

(7) Reactor Cavity Cooling

In the PFR the British have provided two stages of last ditch cooling after the DBA excursion. The first of the last ditch cooling systems is a natural circulation air cooling system which supplies cool outside air to a sump located immediately below the secondary containment vessel in the reactor cavity. The cool outside air circulates upward in the cylindrical annulus formed by the concrete wall of the reactor cavity. The upward moving air extracts heat from the secondary containment vessel surface and the heated air is collected near the top of the secondary containment vessel for discharge to the outside via a stack. In the event of a failure of the last ditch air cooling system or a heat

generation in the reactor cavity exceeding the capacity of the air cooling system, (as might be obtained by a reconsolidation of the reactor fuel into a critical mass following the DBA), an ultimate last ditch cooling system is provided by flooding of the cooling annulus with sea water to permit boiling heat transfer for air indefinite period. Both of these last ditch cooling systems depend upon the integrity of the secondary containment vessel under all conditions of DBA. The British are confident that they can contain the DBA within the primary coolant tank and that the additional protection provided by light weight and structural concrete on the inside surface of the secondary containment vessel will assure that the secondary containment vessel will not be breached even in the unlikely event of a primary tank failure.

(8) Relation of Development to Design Effort

At Risley, the British presented a strong case for their unified organizational approach to the functions of research, development, design and construction. Their thesis states that these functions must be under a single management direction if effective results are obtained. The British are particularly proud of their organizational continuity in the fields of development and design. In their view, the success of the PFR is assured by the history of success of the organization. The same design organization has designed all of the major reactor facilities, thermal and fast, in Great Britain.

(9) Sodium Hydraulic Development

At Risley the sodium hydraulic development is focused upon three

general areas; vibration testing, investigation of gas entrainment phenomena, and development of sodium sampling techniques for reactor fuel subassemblies.

- a. Vibration Studies - Vibration studies are being performed upon the control rod guide tube as a structural cantilever supported from the rotating plug and submerged in the sodium pool. Since the control rod guide tube is exposed to the high velocity sodium discharged from the core, there is a potential for damaging vibrations in the guide tube or in the control rod extension within the guide tube. The British did not seem particularly concerned about vibrations of small amplitude (fretting) at the coolant inlet to either the fuel subassembly or the control rod. Concern was expressed for vibration effects upon the tubes of the intermediate heat exchanger; development work was directed to the solution of IHX tube vibration problems.
- b. Gas Entrainment Studies - The British are highly concerned about the potential gas entrainment at sodium-cover gas interface above the reactor core. They have conducted many experiments to predict and confirm gas entrainment above the core with a variety of control devices such as baffles, rods and deflectors. The British propose to install a baffle plate attached to the lower ends of the control rod guide tubes to suppress the swell of the core coolant jet above the core and to direct the coolant flow laterally toward the IHX's. Their experimental work confirms that the baffle plate will effectively reduce surface turbulence of the core coolant jet. An uncertainty exists as to the gas entrainment

effect of the control rod extensions through the sodium-cover gas interface above the core. It is not known whether the guide tubes will suppress or increase gas entrainment. A potential source of gas entrainment also has been identified at the IHX where the heat transfer tubes penetrate the sodium surface.

- c. Reactor Coolant Sampling - Sodium hydraulic studies of coolant sampling techniques have also been undertaken to support PFR design. In the PFR, individual samples are taken from fuel subassemblies by sample tubes located at the outlet of each fuel subassembly and mounted on the control rod shroud tube. The samples are routed out of the reactor vessel via individual tubes for sampling either as a representative mix or as individual streams. The development work has been directed toward transit time studies to ascertain the preferred velocity and flow volume for individual samplers, and to obtain data to measure the effectiveness of mixed sample flows for fuel failure detection.

(10) Reactor System Mockups

The British listed their PFR mockups as follows:

1. A full scale control rod tube mockup for water hydraulic testing.
2. A 1/5 scale model of the reactor core for water hydraulic testing.
3. A full scale 90° sector of the IHX for water hydraulic testing.
4. A 1/7 scale reactor core mockup for gas entrainment studies.

(Primary goal is to establish preferred baffle shape and location for minimum gas entrainment.)

5. A 5/8 scale model of one sector or "leaf" of the reactor outlet flow baffle with two installed IHX's. (This model is intended to develop preferred inlet designs for the dual IHX arrangement within the flow baffle.)
6. A full scale mockup of a reactor core subassembly bundle of one control rod and six fuel subassemblies. The mockup is for water-hydraulic testing.

(11) Instrument

- a. Core Coolant Temperature - Electrical signals will be obtained from all core positions (thermocouple in each fuel subassembly sodium outlet stream). No connectors are provided for the thermocouple leads in-core or in the sodium vapor region. All signals will be carried via MI cable in guide conduits.
- b. Fuel Failure Detection - Liquid sodium samples (for fuel failure detector) will be extracted from the mixed outlet flow from each fuel subassembly. Although the normal sample will be the mixed stream from a cluster of six fuel subassemblies, the PFR design provides special valves to permit selective sampling of each fuel channel.
- c. In-reactor Flow Measurement
 - (1) The drift of the temperature coefficient with time for permanent magnets apparently limits their use when submerged in sodium at high temperatures. The British plan to use the "temperature noise" transit time method for flow velocity measurement, or a variant where a characteristic "charge" is implanted in the flowing stream for downstream detection.

- (2) The British are not interested in turbine flowmeters for sodium. They gave no indication that the mechanical meters will be used at any place in any system.
- d. Sodium Level Control - The sodium level will be controlled for the PFR by a capacitance probe mounted in a vertical dry well. The PFR design group has not decided yet upon "feed-bleed" or other liquid level control methods. In fact, they are relatively unconcerned since the "tank type" primary system does not appear to have critical liquid level control requirements. An unlikely but possible liquid level control requirement may arise if the sodium volume in the system must be changed with power to obtain optimum IHX sodium pressures during operation or to obtain preferred sodium submergence of the core during shutdown (for refueling).
- e. Sodium Leak Detectors - Sodium leaks are best detected by a "spark plug" device in a local leak collection point. Consideration should be given to identification of leak location by appropriate zoning of collection points.
- f. Fuel Handling Equipment Position Sensors - Fuel handling equipment in the PFR is manually controlled although all shield plug indexing motions are driven by electro-mechanical positioners.

MOL-DONK, BELGIUM, December 12, 1966

Persons visited:

Mr. K. J. Bobin, Operational Supervisor
Dr. Bodnarescu, Plant Relations-Testing
Mr. Jacobs, Operations Manager

Time available for the visit to BR-2 thermal reactor at MOL was severely limited by weather and transportation problems. Because of the press of time, activities at MOL were confined essentially to discussions of BR-2 experience with Mr. Bobin and a short visit to refueling canal of the BR-2.

Observations and Data

1. The core subassembly maximum inclination in the BR-2 is 10° from the vertical. The core lattice is 4" center-to-center at the intercept plane of the skew geometry. The overall reactor height is about 30 feet with 7 foot diameter covers at the top and bottom of the reactor vessel. The reactor vessel has an hourglass shape to provide close spacing of experimental beam tubes to the core; fuel assemblies are about 20 feet in length.
2. It has been observed that the reactor vessel ends undergo measureable distortion under 15 atmospheres internal pressure and reactor operating temperatures. The distortion has not caused either operational or safeguards difficulties.
3. It is expected that the replacement of the machined moderator pieces of beryllium forming the reactor core structure will be an extremely difficult problem at some future time. Although the original machining problem on the beryllium logs was complex, procurement and installation of beryllium moderator caused no significant difficulty.
4. The BR-2 is refueled under water by use of hoists and manual grapples. No special fueling machine is used and the fuel assemblies are not reinforced by stiffeners or other support. The technique used for refueling is to move a hanging (plumb) fuel assembly over the lattice position to be refueled, lower the fuel assembly until the nose of the

assembly enters the lattice position, move the hoist head in the direction of the skew to align the fuel assembly with the skew geometry, and finally, lower the fuel assembly into the core by the hoist. No operational problems have developed using this technique for driver fuel assemblies.

5. Test assemblies with heavy weights at the top of the assembly (instrument connectors, coolant connectors, or mechanical manipulators) may cause charge-discharge problems. Unless the hoisting operation described for driver fuel assembly refueling is performed with care, it is possible to apply excessive bending loads to the test assembly during the charging process. An example was cited where a test assembly with a retractable specimen (actuated by a hydraulic piston and cylinder at the upper end of the test assembly) was bent during charging when the crane operator failed to maintain alignment of the hoist cable with the skew geometry.
6. In general, BR-2 test specimens are mounted in axial, vertical position in the test assembly carrier. The test carriers are generally stronger and stiffer than the normal fuel assemblies. The typical experimental test train in the BR-2 is 3/4 inches in diameter and 15 feet long.
7. No particular problems have been encountered with respect to galling, abrasion or fretting of fuel or test assemblies mounted in the BR-2.
8. The BR-2 operations personnel report that there has been no evidence in test or driver fuel performance that non-verticality of core subassemblies contributes detrimentally (or beneficially) to the performance of nuclear fuel. In their recollection there was just one experiment in which non-verticality was a significant parameter. The test in question was a mechanical test in a subassembly position at the periphery of the core.

The specimen consisted of a rotating journal supported by balls within the test carrier. The unbalanced lateral loading at a 10° tilt was sufficient to cause damage to the rotating elements as a result of excessive mechanical loading.

KARLSRUHE, GERMANY, December 13, 1966

Persons visited:

Professor D. Smidt

Mr. Muller

Mr. Schlechtendahl

Mr. Gast

Mr. Bojarski, Irradiation Facilities

Dr. Kummerer, Fuel Development

Dr. Bohm, Metallurgy

The one-day visit to Karlsruhe was quite short because of the need to drive by auto to and from Stuttgart on December 13. The discussions were conducted with the liquid metal (sodium) cooled fast breeder advocates at Karlsruhe instead of the steam cooled counterparts. The conceptual design work for the Karlsruhe prototype LMFBR has not progressed very far. The Germans appear to be watching for mistakes of others as they pursue their careful conceptual design course. Their present progress and general cautious attitude is in contrast to their technical enthusiasm for the LMFBR. Mr. Gast had only recently returned from a two-year exchange employment with the General Electric Company at San Jose where he had worked on the SEFOR reactor. For this reason, perhaps, it appeared much of the conceptual evaluations being performed by the Germans used the reactor concepts favored by GE as reference position.

Observations and Notes

1. The Germans plan to construct a 1/3 scale model of the reactor core and possibly the primary tank for water hydraulic testing.

SACLAY, FRANCE, December 14, 1966

The visit to Saclay was in the company of Mr. Ken Horton, USAEC Headquarters; Dr. Ed Zabroski, General Electric Company, San Jose; and Mr. Fred Smith, Argonne National Laboratory.

The discussions at Saclay dealt with fuel pin design and material characteristics of reactor fuels and did not progress to the features of the Phenix conceptual design. Details of the fuels and materials discussions are adequately reported by the other participants in these discussions.

During the short visit to Saclay, a sodium laboratory tour was provided for the specific purpose of demonstrating the latest French technique for obtaining uncontaminated sodium samples for analysis. The technique appears to be useful only in laboratory applications although, with ingenuity, it might be applied to routine service operations on a reactor. In principle, the technique requires the temporary diversion of a continuously circulating sodium stream in a test loop to provide a sample in a glass container which may be separated from the loop by successive operations of valving, evacuation and fusing of a separable glass neck. After separation, the sample consists of a sealed, evacuated, spherical glass bottle containing a sodium sample. The sample is kept in the glass bottle until it is removed by fracturing the bottle at the time of examination. The French claim to have obtained excellent results in sodium analysis using the sampling technique described. In particular, they claim to have made a significant improvement in their ability to reproduce and confirm analyses from a single source through multiple samples.

CADARACHE, FRANCE - December 15 and 16, 1966

Persons visited:

Georges Venderyes, Head Fast Neutron Department

Guy Denielou, Assistant Head Fast Neutron Department

Louis Vautrey, Assistant Head Fast Neutron Department

Georges Cajac, Rapsodie Operations Manager

Jean Villeneuve, Head Technology and Heat Transfer Service

Jacques Gollion, Head Mechanical Test Program for Phenix

Jean Leduc, Head Group charged with Title I studies for Phenix

Noil Lions, In charge of Sodium Water Reaction Studies

The morning of December 15 was spent discussing the Rapsodie reactor including some of the more significant design features. Rapsodie itself was visited in the latter part of the morning during a tour conducted by Denielou. During the afternoon of December 15, the current design status of the Phenix reactor and its conceptual features were discussed by members of the Phenix design staff. At the end of the day, the sodium and water mockups for both Rapsodie and Phenix were visited. Messrs. Horton, Zebroski and Smith were present at Cadarache on December 15 for the initial discussions of the Rapsodie reactor. They returned to Paris on the evening of December 15, hence they were not present for discussions on December 16 at Cadarache.

On the morning of December 16, some of the troubles that had been encountered during the pre-startup operations on Rapsodie were discussed by Denielou. The Rapsodie discussion was followed by a discussion of conceptual choices for the Phenix reactor. During the afternoon of December 16, discussions were held on the conceptual problems of the FFTF and current FFTF activities. The visit at Cadarache ended with a final trip through the Rapsodie reactor to observe sub-pile regions which had not been visited the previous day.

Observations and Notes

1. Problems encountered during pre-startup operations with Rapsodie:

a. Reactor Vessel Distortion

Denielou described the problem of Rapsodie reactor vessel distortion and the corrective measures that have been applied. The Rapsodie reactor vessel has the shape of a cylinder with a conical bottom terminating in the single coolant inlet pipe. The inlet pipe connects concentrically to the bottom of the conical section of the vessel and makes a "U" bend vertically upward along the side of the vessel before connecting to a horizontal pipe segment which penetrates the reactor cavity above the core. In effect, the vessel and bent inlet pipe form a trap to assure that sodium coolant does not drop below the level of the core in the event of leakage failure of piping, pumps, or heat exchangers outside the reactor cavity. Unfortunately, the single coolant inlet pipe in its bent configuration creates an asymmetric structural system. During sodium charging of the vessel and primary circuit and subsequent heat tests, a measurable distortion of the vessel was observed. The distortion resulted in a lateral displacement (about 16 mm) of the lower end of the reactor vessel at the inlet pipe. Measurements from the top of the reactor vessel through control rod shrouds confirmed that the vessel had distorted and that the lateral displacement at the core was about 8 to 10 millimeters. The measured displacement of the vessel was considered to be excessive and corrective measures were applied. The hypothesis developed to explain the vessel distortion was that a nonuniform

circumferential temperature distribution on the reactor vessel shell caused by the asymmetric heating (or cooling) effect of the single inlet coolant pipe caused the thermal distortion. Corrective measures consisted of the installation of gas baffles about the inlet coolant pipe and the installation of additional insulation on the upper courses of the reactor vessel. The effect of the corrective measures was to reduce the deflection of the reactor vessel at the inlet pipe connection to 8 - 10 mm. These measures resulted in a core displacement by thermal distortion of less than 6 mm., which was deemed acceptable.

b. Cover Gas Overpressurization

Rapsodie had recently undergone an embarrassing and messy accident which had been foreseen but not adequately provided for in design. In brief, because of an operator error with the argon cover gas control system and a deficient design for the control rod shrouds, sodium was forced up the control rod shroud tubes to overflow onto the reactor cover to a depth of about two feet. The sodium thereupon froze and had to be removed by a tedious digging process. To explain how this accident happened, it is necessary to first describe part of the reactor vessel and cover gas system arrangement before describing the operation that resulted in the sodium overflow. The reactor vessel and primary circuit are connected to a dump tank which will contain the entire contents of the primary sodium system. The dump tank is provided with both liquid sodium and cover gas pipe connections to the reactor vessel to permit interchange of both gas and sodium during a system

dump or fill. These connections are large size pipes equipped with block valves. The cover gas pressure control and purification system is connected to the large pipe joining the gas space in the reactor vessel and dump tank. The flow capacity of the cover gas pressure control and purification system is quite limited; that flow rate formed the design basis for the reactor vessel pressure relief valve capacity and also formed the design basis of the size of relief holes in the control rod shroud. The relief holes were above the normal liquid level but below the reactor vessel cover to permit sodium which might be forced up the shroud tube to spill over out of the vent holes back into the reactor vessel before being conducted up the shroud tube and onto the reactor cover. The available flow rate of cover gas from the normal purification circuit was insufficient to accomplish a coolant level change at the desired rate; consequently, to obtain larger flows, a procedure had been developed to open the valve on the gas piping, connecting the dump tank and the reactor vessel. The flow was controlled by intermittent manual operation of the remote controlled gas valve. The valve operation was controlled by a pushbutton in the control room. It is believed the operator became confused and interchanged "the short open cycle and long closed cycle" of the intermittent valve operation with a "long open cycle and a short closed cycle" which resulted in overpressure of the reactor vessel. The gas flow exceeded the capacity of the reactor pressure relieving system and also exceeded the venting capacity of the holes in the shroud tube. As a result of excess cover gas pressure in the reactor vessel, which could

not be relieved by vessel vents, the sodium was forced up the control rod shroud tubes to overflow onto the reactor cover.

c. Primary Pump Bearing Problems

The Rapsodie primary pumps had been operated many times without difficulty during the pre-startup circulating and heat tests. Without apparent cause, the lower hydrostatic sodium bearing recently seized on the journal. The pump was withdrawn and inspected with no cause for the bearing failure identified although the bearing was damaged by the seizure. No hypothesis exists as to why the bearing failed and concern prevails that a failure may occur again after reactor startup. It has been postulated that the bearing casing may have become thermally distorted but there were no unusual operational conditions noted at the time of the bearing failure to confirm this theory. The pump is a free surface design with a hydrostatic journal bearing in a sump-type configuration. Pipe connections to the pump barrel occur in the space above the hydrostatic bearing which allows the possibility of thermal distortion resulting from cold or hot fluid additions to the barrel. When queried, Denielou did not show much concern for gross vessel distortion which may result from the local injection of a hot or cold coolant stream in an otherwise isothermal vessel structure.

d. Air Dump Damper Control Problems

The Rapsodie is similar to the FFTF in its air cooled heat dump on the secondary circuit. Apparently Rapsodie has had continuing problems with the structural and mechanical deficiencies of the control louvers on the heat dump.

The identified deficiencies are:

1. Faulty seals on damper blades
2. Binding and seizure of bearings and linkages for damper leaves.
3. Damper control characteristics are unstable at low flow conditions.

e. Automatic Reactor Control

The Rapsodie has an automatic control system for nuclear control. The system has been demonstrated to be unstable and nonpredictable during pre-startup tests. Consequently, it will not be used to provide initial hands-off automatic nuclear control for the reactor. During the course of pre-startup tests on Rapsodie, a stringent effort has been made to reduce the number of instrument signals and control functions that will either prevent the reactor operation or initiate a scram. The control room shows numerous blank panels where instrument devices have been removed.

f. Control and Safety Rod

The control and safety rods for the Rapsodie had been continuing problems. They are precision devices with spring-driven scram accelerators. Each unit represents a selective assembly and an individual development program before it can be certified for installation in the reactor. The problems run the gamut of bellows integrity, seal leakage, shaft and sleeve misalignment, galling of mating parts, poor position reproducibility, deficient scram acceleration, and deceleration snubber development. It was stated that although the present control and safety rods are usable and safe after the extensive individual

development treatment, new designs for the rods will be developed. The dominant criteria for new rods will be simplicity and reliability in both assembly and operation. The significant goal will be the removal of all close fitting alignment guides in sodium or sodium-cover gas environment.

g. Rapsodie Mockups

A major sodium mockup is provided for developmental support of Rapsodie. A full scale reactor vessel mockup (HR-1) is connected to a 10 MW and a 1 MW sodium circulating loops. Mockup was completed in 1961 and the loops were completed in 1962. The 10 MW loop is prototypical of each of the two 10 MWt loops in Rapsodie. Loops and mockup were fabricated of type 321 stainless steel (Rapsodie material is type 316 stainless steel). HR-1 loops were provided with temperature transient capability of 100°C/second by rapid transfer of stored sodium. The HR-1 loops and mockup served as a test stand for the Rapsodie fuel handling machine.

In building HR-2, hydraulic equipment originally used for Rapsodie tests are now used for testing of Phenix fuel subassemblies. The HR-2 loop is not a thermal-hydraulic facility.

The French reported good utility from their mockups. However, they claimed to have established needed correlations with predictive analysis so that full scale sodium mockups will not be needed for Phenix. The only apparent reservation to this approach rests in the performance of Rapsodie which is due to startup early in 1967.

h. The Phenix Reactor

The reactor power of 250 MWe was selected for Phenix over the earlier planned 125 MWe rating because the higher power offers more prototypic apparatus and reactor performance. A reactor greater than 250 MWe power is limited by the availability of French plutonium by 1972.

The Phenix will be a tank type LMFBR in which the reactor and all primary coolant components are suspended in a large (about 40 feet diameter) tank. The French are concerned about the structural integrity of the tank wall at the sodium-cover gas interface. To avoid catastrophic failure of the tank in the event of thermal cracking to the tank wall at the interface region, the French propose to use suspension rods which attach to the tank wall below the sodium-cover gas interface. The suspension rod attachment points to the tank wall will also serve as the support points for the reactor core structure.

The French felt that the development requirements for a fuel handling system similar to the British PFR would be too risky both in time and expected performance. Therefore, instead of the pantograph fuel manipulator and rotating plugs at the reactor cover for refueling, the Phenix will have an ingenious ramp conveyer to withdraw the spent fuel (in sodium filled cask) through the reactor cover and out through the containment vessel fuel lock.

Control rod design for Phenix is not established. Strong design emphasis will be placed upon the Rapsodie findings; namely, simplicity and reliability cannot be overemphasized.

The Phenix IHX design concept is established but the design is not complete. The IHX is a suspended unit in the primary tank with primary coolant inside of tubes and the secondary coolant in the shell side. A crude valve will be provided for temporary shutdown of an IHX. The valve consists simply of a disc which is lowered onto the tube sheet to seal off primary coolant flow.

The Phenix pump will be rated at about 4000 cubic meters/hour and 8 bars. Difficulty was observed in the conflicting design requirements for a short pump shaft and a submerged pump section. An adjustable pump inlet shroud is provided to assure submergence of the pump inlet during all operational modes.

The Phenix core consists of 120 fuel hexagonal subassemblies in the core each measuring 12 centimeters across the flats. The active core height to diameter ratio is about 0.5 ($\frac{0.7 \text{ meters}}{1.6 \text{ meters}}$).

i. Identified Documents

CEA-R2583	H ₂ and O ₂ Impurities in Liquid Sodium	<u>Issued</u> 1964
CEA-2369	Deposition of Sodium Impurities in Coolant Circuits - Dosage Observed	1963
CEA-835	O ₂ Dosage in Sodium	
CEA-R3028	Corrosion of Niobium - Circulating Sodium at 600°C	1958

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December 6, 1966	" "
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December 16, 1966	" "

APPENDIX II

ORGANIZATIONAL CHART

DOUNREAY EXPERIMENTAL REACTOR ESTABLISHMENT

DOUNREAY EXPERIMENTAL REACTOR ESTABLISHMENT

APPENDIX III

RISLEY DATA SHEETS

SODIUM HEAT TRANSFER

DATA SHEETLIQUID METAL HEAT TRANSFER COEFFICIENTS TO BE USED FOR P.F.R. DESIGNS

For calculating the heat transfer film coefficient of the liquid metal when designing heat transfer or heat removal systems for P.F.R. for heat removal or transfer equipment used in the normal operation of the reactor the area as determined by the appropriate equation only has to be included with no percentage allowance.

For emergency or safety circuits such as thermal syphon systems, it may be advisable to add a factor to the determined area. The amount so added will be determined by assessing the actual area based upon the equation, and the probable temperatures if an increased area is included.

LIST OF SYMBOLS

(A) Flow Through Circular Tubes		(C) Parallel Flow Through Shell of a Shell-and-Tube or Rod Heat Exchanger	
Nu	Nusselt Number = h^D/k (Dimensionless)	Nu	Nusselt Number = $\bar{h}^{Deg}/k$ (Dimensionless)
h	Coefficient of Heat Transfer Between Liquid Metal and Tube Wall, $CHU/ft^2hr\ ^\circ C$	$\bar{h}$	Mean Coefficient of Heat Transfer Between Fluid and Rod, $CHU/ft^2hr\ ^\circ C$
D	Inside Diameter of Tube, ft	Deg	Hydraulic Mean Diameter of Flow Passage = $\frac{4 \times \text{Total Wetted Area}}{\text{Total Wetted Perimeter}}$ ft
k	Thermal Conductivity of Liquid Metal, $CHU/ft\ hr\ ^\circ C$	P_e	Peclet Number = $R_e \times Pr$
P_e	Peclet Number = $R_e \times Pr$ (Dimensionless)	R_e	Reynolds Number = $\rho V^{Deg}/\mu$ (Dimensionless)
R_e	Reynolds Number = $\rho VD/\mu$ (Dimensionless)	P	Pitch Between Rod Centres, ft
ρ	Density of Liquid Metal, lb/ft^3	D	Diameter of Tube or Rod, ft
V	Fluid Flow Velocity, ft/hr	(D) Flow Across a Staggered Tube Bank	
μ	Fluid Absolute Viscosity, $lb/ft\ hr$		
Pr	Prandtl Number = C_p/k (Dimensionless)		
C	Specific Heat of Fluid, $CHU/lb\ ^\circ C$		
(B) Flow Through Non-Circular Tubes and Annuli		Nu _{av}	Average Nusselt Number = $\bar{h}^D/k$ (Dimensionless)
Nu	Nusselt Number = $\bar{h}^{Deg}/k$ (Dimensionless)	$\bar{h}$	Mean Coefficient of Heat Transfer Between Liquid Metal and Tube Wall, Averaged Around Periphery of Tube and Is Equal to $3.45 (R_{ef})^{0.66} CHU/ft^2hr\ ^\circ C$
$\bar{h}$	Mean Coefficient of Heat Transfer Between Fluid and Tube Wall, Defined in Terms of the Arithmetic Mean Heat Flux and the Arithmetic Mean Wall Temperature, Both Averaged Around the Periphery of the Duct, $CHU/ft^2hr\ ^\circ C$	D	Diameter of Tube, ft
Deg	Hydraulic Mean Diameter of Non-Circular Tube (or Annulus) = $\frac{4 \times \text{Total Wetted Area of Tube}}{\text{Total Wetted Perimeter of Tube}}$ ft (Total Wetted Perimeter of Tube Being Considered Irrespective of Whether There is Heat Flow Across the Whole of the Periphery or Not) = $2(r_2 - r_1)$ for an Annulus, ft	R_{ef}	Reynolds Number with Fluid Properties Evaluated at Mean Film Temperature $R_{ef} = \rho f V_{max}^D/\mu_f$ (Dimensionless)
Nu _s	Nusselt Number for Slug Flow Through the Tube or Annulus in Question (Dimensionless)	V_{max}	Velocity of Fluid at Narrowest Cross-Section in Tube Bank, ft/hr
P_e	Peclet Number = $R_e \times Pr$	ρf	Fluid Density at Mean Film Temperature, lb/ft^3
R_e	Reynolds Number = $\rho V^{Deg}/\mu$ (Dimensionless)	μ_f	Viscosity at Fluid Mean Film Temperature, $lb/ft\ hr$
r_1	Inner Radius of Annulus, ft	P_e	Peclet Number Based on V_{max} with Fluid Properties Evaluated at Bulk Temperature. $P_e = R_e \times Pr$ (Dimensionless)
r_2	Outer Radius of Annulus, ft	R_e	Reynolds Number = $\rho V_{max}^D/\mu$ (Dimensionless)
		Physical Properties of Fluid Evaluated at Bulk Temperature, Except in the Specific Cases of μ_f and ρf in R_{ef} .	

CONFIGURATION	EQUATION TO BE USED	REMARKS
<p>Flow Through Circular Tube; Constant Wall Heat Flux</p>	$Nu = 5.0 + 0.025 p_e^{0.8}$	<p>P_e Range, 40-20,000 No Contact Thermal Resistance at Tube Surface.</p>
<p>Flow Through Tubes of Noncircular Cross-section</p>	$Nu = 2/3 Nu_s + 0.025 p_e^{0.8}$	<p>No Contact Thermal Resistance at Tube Surface. Values of Nu_s for Different Shapes Given in Table No. 1.</p>
<p>Flow Through an Annulus with Heat Flow Across One Diameter Only. Constant Heat Flux</p>	$Nu = 0.75(r_2/r_1)^{0.3}(7 + 0.025 p_e^{0.8})$	<p>No Contact Thermal Resistance at Duct Wall. Valid for Heat Flow Across Either Inner or Outer Diameter When r_2/r_1 Is Less Than 1.4. Valid for Heat Flow Across Inner Diameter Only When r_2/r_1 Is Greater Than 1.4 but Less Than 10.</p>
<p>Parallel Flow Through Shell Side of a Shell and Tube Type Heat Exchanger or Through the Space Between a Bundle of Rods</p>	<p>Triangular Pitched Tubes:</p> $Nu = 7.0 + 3.8(P/D)^{1.52} + 0.027(P/D)^{0.27} p_e^{0.8}$ <p>Square Pitched Tubes:</p> $Nu = 7.0 + 4.24(P/D)^{1.52} + 0.0275(P/D)^{0.27} p_e^{0.8}$	<p>P/D Range 1.375 to 10. This Gives the Theoretical Values Only and a Correction Factor As Shown in Figure No. 1 Must Be Applied.</p>
<p>Flow Across a Staggered Tube Bank, for Example a Shell and Tube Exchanger Baffled to Give Effective Cross Flow</p>	$Nu_{av} = p_e^{0.5}$	<p>P_e Range, 50-40,000</p>

TABLE NO. 1	
Boundary Condition:- Heat Input Per Unit Length Constant and Wall Temperature Constant Around Periphery of the Duct at a Given Axial Position.	
Geometry	Nu_s
Circle	7.5
Square	7.03
Equilateral Triangle	6.67
Infinite Slot	12.0
Infinite Slot with One Wall Insulated	6.0
Ninety Degree Isosceles Triangle	6.55

RELATIONSHIP BETWEEN Pe & Nu No's FOR SHELL - SIDE
FLOW IN PRI/SEC FAST REACTOR HEAT EXCHANGERS

RELATIONSHIP BETWEEN Pe & Nu No's FOR SHELL - SIDE
FLOW IN PRI/SEC FAST REACTOR HEAT EXCHANGERS.

RELATIONSHIP BETWEEN Pe & Nu NO'S FOR TUBE -
SIDE FLOW IN PRI/SEC. FAST REACTOR HEAT EXCHANGERS

FIGURE NO. 5

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