# DEVELOPMENT OF A NEW VVER-440 FUEL DESIGN

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## ABSTRACT

In March 1996 BNFL signed a contract with Imatran Voima Oy (IVO) and PAKS Nuclear Power Plant to design, develop, license and supply 5 Lead Test Assemblies to the VVER-440 reactor at Loviisa in Finland.

In June 1998 the manufacture of these 5 assemblies (4 fixed assemblies and 1 follower assembly) was completed. The fuel is expected to be loaded into Loviisa's Unit 2 reactor during the shutdown scheduled for September of this year.

This paper describes the development programme carried out in support of the new design. It describes the work carried out by BNFL immediately after contract signature to optimise the key parameters of the design (e.g. fuel rod diameter, fuel rod pitch, etc.). It gives an overview of the thermal hydraulic testwork undertaken to establish critical heat flux limits for the new fuel design and to demonstrate pressure drop compatibility with existing fuel. It describes the series of mechanical tests carried out to demonstrate that the fuel is sufficiently strong to withstand the maximum loads expected during transport, handling and reactor operation, including both normal operation and fault conditions. Finally it describes the analytical work undertaken using BNFL methodology to assess that relevant design criteria are met.

The results of the development programme have been incorporated into a Final Design Report which forms the basis of the licence application to load the LTAs at Loviisa. It will also form the basis of future applications to import reload quantities into both Finland and Hungary.

# 1. NOVA E-3 DESIGN

## 1.1. Design Concept

On contract signature it was agreed that the new fuel design would have a removable shroud with a reduced thickness of 1.6 mm and that Zircaloy-4 material would be used for all spacer grids, shroud tubes and fuel tubes. It was also agreed that it would be desirable for the assembly to be reconstitutible, i.e. for the fuel rods to be removable.

BNFL then embarked on an exercise to optimise the key parameters of the fuel design. For the purposes of the exercise the following parameters were considered as variables; number of fuel/water rods, fuel rod pitch, fuel rod diameter and pellet geometry. Six design variants were examined from the point of view of fuel cycle costs, neutronic and thermal-hydraulics behaviour, spent fuel pond subcriticality and mechanical strength. The investigations were supported by calculations carried out by IVO and PAKS. The design variant, identified as NOVA E-3, was chosen as optimising fuel cycle costs, a relatively high uranium mass (to keep burnups low) and low surface heat flux (to maintain large margins to critical heat flux) and adequate mechanical strength. The key features of the NOVA E-3 design are listed below:

| | |
|---|---|
| Number of fuel rods | 126 |
| Number of water rods | 0 |
| Fuel rod pitch | 12.28 mm |
| Fuel rod diameter | 8.9 mm |
| Shroud thickness | 1.6 mm |
| Pellet geometry | Solid |

The exercise also highlighted that, for PAKS at least, all the design variants considered would require the use of burnable poisons to maintain acceptable subcriticality in the PAKS spent fuel ponds. For PAKS therefore it was envisaged that a number of gadolinia doped rods would be included in any fuel supplied to them.

## 1.2. Design Details

The development programme described in the following sections of this paper confirmed the feasibility of the above design concept and established, sometimes after feedback from tests or analysis, further details of the design.

The fixed fuel (Fig 1) and follower fuel assemblies consist of stainless steel top and bottom nozzles, a Zircaloy-4 shroud tube, a fuel rod bundle, and a Zircaloy-4 central instrumentation tube. The fuel rod bundle consists of 126 Zircaloy-4 clad fuel rods spaced on a triangular pitch within an hexagonal array, and is supported by Zircaloy-4 spacer grids. Fuel rods are laterally and axially restrained at each grid by means of a spring reacting on two dimples in each grid cell, and are located in the bottom flow plate by a split bottom end plug. The bottom flow plate is attached to the bottom nozzle by a support rim. A top flow plate is provided to prevent ejection of the rods from the assembly and is attached to the top nozzle by support pins. The grids are located on the central instrumentation tube by means of expansion joints, and the instrumentation tube is attached to the bottom nozzle by an instrumentation tube screw. The shroud tube is attached to both nozzles by screws at each end, which allow the shroud to be removed from irradiated assemblies for inspection purposes. The key fuel assembly and rod parameters are listed in Table 1. Material specifications are chosen to comply with the principles of the dose rate reduction philosophy, known as 'As Low As Reasonably Practicable (ALARP)'.

## 2. TEST PROGRAMMES

### 2.1. Thermal-Hydraulics Tests

The NOVA E-3 thermal-hydraulics tests were performed at Columbia University's (CU's) Heat Transfer Research Facility (HTRF) in New York. The tests addressed fuel assembly pressure drop, critical heat flux, subchannel mixing, flow stability and flow endurance. The HTRF has widely recognized experience in the field of reactor thermal-hydraulics, having completed test programmes on over 350 rod bundles since the mid-1960s [1-4]. Data gathered includes critical heat flux, isothermal pressure drop and subchannel mixing data. The HTRF has a closed circuit steam water loop with a power supply of up to 13 MWe (see Fig.2). The HTRF is fully instrumented with advanced and reliable measurement instrumentation. Data is collected by a high speed, computer controlled data acquisition system.

#### 2.1.1. Pressure Drop Tests

Pressure drop tests on both fixed and follower fuel assemblies were undertaken. The objective of these tests was to evaluate the pressure drop characteristics of the fixed and follower fuel assemblies. A schematic drawing of the pressure test instrumentation arrangement is shown in Fig. 3. The pressure tappings on the shroud tubes of the test assemblies were positioned to establish the frictional losses and the losses of the assembly internal components. Tapping points were also introduced in the inlet and outlet manifolds to give the overall pressure drop across both fixed and follower assemblies.

The test rig was pressurised to the normal reactor operating pressure of 12.2 MPa and static pressure measurements were recorded at several temperatures between room temperature and 270°C, and at several flow rates at each temperature over the range of normal operation including the pump overspeed condition. At each of the test conditions five repeat measurements were recorded for each tapping point, resulting in over 4000 pressure drop datapoints for each of the two fuel assemblies.

Regression analysis of the pressure drop data produced loss coefficient-Reynold's number correlations for each component and for the fixed and follower assembly overall losses.

### 2.1.2. Critical Heat Flux Tests

In order to evaluate the Critical Heat Flux (CHF) characteristics of the BNFL NOVA E-3 assemblies, tests were performed on three different rod bundles, each with a different radial power distribution. A schematic drawing of the CHF test section is shown in Fig. 4. Each bundle contained 19 electrically heated rods arranged on a triangular pitch in an hexagonal array. The rod outer diameter, rod pitch and heated length were identical to the NOVA E-3 assembly values. The rod bundle was mounted vertically in the rig and was surrounded by a ceramic liner simulating the assembly shroud tube. The bundle was supported by spacer grids positioned on the same pitch as in the NOVA E-3 design. Three directional thermocouples (T/Cs) were located on each rod just below the end of the heated length to act as the primary detectors of Departure from Nucleate Boiling (DNB). The uppermost grid was axially positioned such that the bottom of the grid was exactly one grid pitch below the end of the heated length in order to encourage DNB to occur close to the end of the heated length, where the rod directional T/Cs were located. An additional support grid was positioned above the end of the heated length to locate an array of 36 thermocouples which measured the subchannel exit temperatures. The rod and subchannel radial geometry and numbering applied in the tests and the T/C arrangement are illustrated in Fig. 5.

The first test was performed with a uniform radial power distribution and the second and third tests with different non-uniform radial power distributions. In total 230 CHF datapoints were generated over a wide range of fluid pressures, mass velocities and qualities.

Isolated subchannel analyses were performed using the Smolin [5,6], Bezrukov [7], and other correlations in order to predict CHF values for comparison with the measured data. In the Bezrukov correlation analysis, 229 of the 230 CHF datapoints gave predicted/measured (P/M) values less than the correlation limit, i.e. 99.6% of the P/M values are bounded by the limit. In the Smolin correlation analysis, 228 of the 230 datapoints gave P/M values less than the correlation limit, i.e. 99.1% of the P/M values are bounded by the limit. Furthermore in isolated subchannel analyses of in-reactor fuel a subchannel enthalpy rise uncertainty factor is conservatively applied to the hot subchannel enthalpy rise. Thus, it is concluded that the existing DNB methodology continues to be acceptable for use with NOVA E-3 fuel.

Subchannel analyses using the VIPRE code [8] and the Bezrukov, Smolin and other correlations were initially performed to assess the applicability of the correlations in subchannel analysis of NOVA E-3 fuel, with a view to creating modified versions of the correlations that would give more accurate CHF predictions. The Bezrukov and Smolin correlations behaved similarly, generally giving predicted CHF values larger than the measured values, and there were noticeable trends in the P/M values with inlet mass velocity, exit pressure and local quality. Because of the relatively simple form of the Bezrukov correlation it was decided to re-optimise this correlation for use in subchannel analysis. Thus the six coefficients in the correlation were varied until an optimised form of the correlation was arrived at which gave a mean P/M value of one and a minimum standard deviation in P/M. The form of the correlation is as follows:

$$\dot{q}_{cr} = a_1 (1 - x)^{a_2 + a_3 p} G^{a_4 + a_5 (1 - x)} (1 + a_6 p)$$

In the local steam quality range -0.15 to 0.23 the P/M values were shown to be normally distributed and it was confirmed that there were negligible trends in the P/M values with inlet mass velocity, exit pressure and local quality. Calculated results with optimised coefficients for the 173 data points in this steam quality range show a mean measured to predicted value of 1.0023 with a standard deviation of 0.0999.

### 2.1.3. Subchannel Mixing Tests

The current method of assessing the possibility of bulk boiling at subchannel exit is to consider the subchannels in isolation. By ignoring fluid exchange and thermal mixing between subchannels this method is conservative. The objective of these tests was to derive a mixing coefficient or Thermal Diffusion Coefficient (TDC) for use in a less conservative and more realistic subchannel analysis of bulk boiling, in which both fluid exchange and thermal mixing between subchannels are modelled. Subchannel exit temperature data were obtained from 52 test runs. The tests were carried out on the CHF test sections (see 2.1.2) and the subchannel exit temperatures were measured by the array of 36 thermocouples at the test section exit.

Each test run was modelled by the VIPRE thermal-hydraulics code. The TDC value implemented in VIPRE was progressively altered until the root-mean-square variation in the measured / predicted subchannel enthalpy rise values was a minimum. The resulting TDC value was then the best estimate value for that run.

Use of the coefficient in exit boiling calculations demonstrates the conservatism inherent in the traditional methodology.

### 2.1.4. Flow Stability Tests

Two flow stability tests were carried out at the HTRF on a full size prototype fixed fuel assembly. The assembly was tested at hot pressurised conditions over a range of flow rates bounding those expected in normal operation. Displacement and acceleration transducers were mounted on the assembly at various locations. The data from these measurements have been used to assess the amplitude of vibration of the fuel bundle as a whole and of individual fuel rods. Five devices were installed on the test assembly: two accelerometers to measure shroud vibration and three eddy current devices to measure either grid or rod vibration. The accelerometers were installed so that if significant shroud vibration occurred its effect could be subtracted from the eddy current device measurements. In the first test the eddy current devices were mounted flush with the shroud tube inside face and aimed at Zircaloy targets on the spacer grids; in the second test the eddy current detectors were repositioned to aim at Zircaloy targets in selected fuel rod simulators.

The results showed that the vibration levels were extremely small. It was concluded that there was no evidence of resonances in either the fuel assembly as a whole, or in the individual fuel rods.

### 2.1.5. Flow Endurance Test

The endurance test was conducted at the HTRF by placing the loop in automatic control and continuously running it for 1000 h at hot, pressurised conditions. Since there was no evidence of flow-induced resonances occurring at any particular flow rate (see 2.1.4), this long term wear test was performed at the nominal assembly flow rate.

The results of the endurance test were acceptable to BNFL and the customer.

## 2.2. Mechanical Tests

### 2.2.1. Component Strength Tests

To ensure that loads encountered during shipping and handling and reactor operation, including faults, do not damage the assembly components or their joints or connections, thus jeopardising their subsequent behaviour in reactor, a series of mechanical tests were carried out by Bodycote Materials Testing Ltd. (UK).

Shroud Strength, Shroud to Nozzle Joint Strength and Fixed Fuel Assembly Nozzle Strength Tests. The aim of these tests was to confirm that the shroud tube, shroud tube to nozzle joints and top and bottom nozzles can withstand the maximum tensile, compressive and torsional static design loads and, in addition, to confirm that the shroud tube to nozzle joints can withstand axial fatigue loads. The tensile, compressive and torsional tests were carried out at ambient and operating temperatures.

Grid Strength Tests. In order to determine grid stiffness under different conditions (shipping, handling and seismic/LOCA loads) and to derive design limit criteria, static and impact tests were carried out at ambient and operating temperature.

Follower Fuel Assembly Top and Bottom Nozzle Strength Tests. The aim of the test was to confirm that the tensile, compressive and torsional strength of the follower fuel assembly top and bottom nozzles are acceptable at the design loads.

Instrument Tube to Grid Sleeve Joint Strength Tests. In order to confirm that the strength of the joint is acceptable at the design loads, the axial deflection of the joint was tested at ambient and at operating temperatures. The specimens comprised a section of the grid sleeve mechanically joined to a section of instrumentation tube.

Fuel Rod/Instrumentation Tube to Bottom Nozzle Joint Strength Tests. Static tensile tests of the joint were carried out at both ambient and operating temperatures to confirm the acceptable strength of the instrumentation tube to bottom nozzle joint at the design loads and that the fuel rod could be removed from the bottom nozzle flow plate at a predetermined load.

Grid Spring Stiffness Tests. The grid springs are required to minimise risk of fuel rod fretting. In practice this is achieved by ensuring (under the worst combination of tolerances and operating conditions) that a positive spring force is maintained. A number of grid straps containing an appropriate number of spring specimens and a section of grid with the grid springs in their cell locations were used in stiffness tests to determine load/deflection curves.

Holddown Coil Spring Stiffness Tests. The holddown springs are required to prevent lift-off of the fixed fuel assembly in normal operation and all anticipated operational occurrences, except pump overspeed. Coil spring specimens were tested at ambient and operating temperatures and the results used to determine the spring stiffness for design calculations input.

Fuel Assembly Vibration Tests. The vibration characteristics of the fixed fuel assembly were determined by measuring: the lateral vibration across the flats and diagonals on the test assembly, the vibration on axial impact with a solid base and the vibration due to lateral impact. In addition tests to determine the axial, lateral and torsional stiffness of the fixed fuel assembly were carried out.

Fuel Rod Vibration Tests. Vibration testing was carried out in order to determine the vibration characteristics of the fuel rod in both air and water.

Follower Fuel Control Assembly Drop Test. In order to prove that the follower fuel assembly design is strong enough to withstand the impact forces generated during a reactor scram and an intermediate rod breakage accident event, controlled and free drop tests were carried out. Both tests were carried out in static water. Examination of the follower fuel assembly showed no evidence of damage in the scram tests, and showed that the follower fuel assembly remains in a condition such that it can be removed from the reactor after a break of an intermediate rod.

Conclusions of the Tests

The tests were carried out using standard mechanical test and measuring equipment and showed that component stresses and deformations were acceptable at the design loads. In addition, data was produced for use in the structural analytical analysis.

### 2.2.2. Materials Tests

All the materials used in the NOVA E-3 designs have been previously irradiated in light water reactors and have accumulated extensive in-reactor service with acceptable material performance. In addition, the cobalt content of major parts is limited in line with the ALARP principle. Advances in the development of light water reactor fuel designs have been incorporated in this design. For example, an optimised cold worked and stress relieved Zircaloy-4 fuel clad that provides a superior combination of low absorption cross section, high strength and good corrosion resistance has been used. The materials used in the design were extensively tested by Bodycote Materials Testing Ltd. and the cladding manufacturer according to BNFL test requirements. These included: mechanical and creep tests, autoclave tests at above 400°C, texture measurements and chemical analysis, fracture toughness, corrosion and fatigue tests. All results from the tests and examinations were acceptable.

The rate of oxidation of the cladding is determined by the temperature and the oxidation conditions in both the coolant and in the oxide film. Cladding oxidation dependence on water chemistry, for most chemistry regimes which are currently in use in PWRs and VVERs, is a second-order effect relative to the effects of power history and coolant temperature. Nevertheless, BNFL expects, on the basis of out-of-reactor tests, that the primary cooling water chemistry in VVER plant will result in a less corrosive environment than the chemistry regimes currently in use in Western PWRs, where Zircaloy-4 cladding has operated successfully for many years. After each cycle of operation of Loviisa, the LTAs will be subject to post irradiation examination (PIE) programme, in which actual clad oxidation thickness will be measured.

# 3. DESIGN CRITERIA

The principal licensing document for BNFL's NOVA E-3 fuel is the Final Design Report. This report presents the design criteria adopted in the areas of thermal-hydraulics design, fuel assembly mechanical design, nuclear design and fuel rod design. The design criteria themselves are chosen to fulfill the requirements of the YVL Guides [9], the American Nuclear Society [10], the ASME [11], the IVO/PAKS/BNFL Contract and BNFL's own guidelines. Demonstration that the criteria are met is based on a combination of the test programmes described above and analytical work. A number of computer codes were used in the analytical work and these are described below.

## 3.1. Design and Licensing Computer Codes

The principal computer codes BNFL has used for thermal-hydraulics design, mechanical, nuclear and fuel rod design calculations are VIPRE, ANSYS, DYNA3D, the CASMO-HEXBU and KARATE systems, and ENIGMA-B. They are described briefly below.

VIPRE. The VIPRE [8] code predicts the three-dimensional velocity, pressure and thermal energy fields for single and two-phase flow in light water reactor cores, during both steady-state and transient conditions. It solves the finite difference equations for mass, energy, and momentum conservation for an interconnected array of channels assuming incompressible, thermally expandable, homogeneous flow. The equations are solved with no time step or channel size restrictions for stability. Although the formulation is homogeneous, a variety of empirical models are included for subcooled boiling and vapour/liquid slip in two-phase flow.

The VIPRE code has been validated against experimental data from a variety of test sections. It is used extensively by utilities within the United States, where it is approved by the US Nuclear Regulatory Commission (USNRC) for PWR licensing calculations. BNFL has used VIPRE in code comparison exercises to give the licensing authorities in the United Kingdom confidence in the thermal-hydraulics codes currently used for thermal analysis of the Sizewell 'B' PWR core.

ANSYS [12] and DYNA3D [13] are general purpose, three-dimensional finite elements programs used extensively in structural analysis. DYNA3D [13] is used to analyse impact related problems. Both codes were used in the mechanical design of the fuel assembly.

CASMO-HEXBU and KARATE. The codes used in nuclear design were the CASMO-HEX, HEXBU-3D (MOD 5), ELSI-1440 system, which have been validated against mathematical benchmarks, critical experiments, and NPP Loviisa operational data [14] and KARATE code system [15]. The CASMO-HEXBU system performs neutron physics calculations using many-group rod-cell models, and two-group single-assembly and full core nodal calculations. It can model both steady state and slow transient conditions. Rod powers are tracked explicitly by the code ELSI-1440. The KARATE code system validation covered a wide range of mathematical and experimental benchmarks.

ENIGMA-B. All the fuel rod design calculations on NOVA E-3 fuel were carried out using the ENIGMA-B 7.3.0 code [16]. The ENIGMA code predicts the behaviour of nuclear fuel rods during in-reactor operation. To do this, the code contains mathematical models which represent the properties of the materials involved and the processes and interactions which take place during irradiation. Its calculations include those for the temperature distribution through the fuel and clad, stresses and deformations in the fuel and clad, fission product release, pellet-clad interaction and clad corrosion. The code incorporates all of the models necessary for performance analysis under normal and off-normal operating conditions. Its numerical schemes are designed to minimise the constraints placed on sub-models and to achieve convergence under all likely conditions of use. The code's fully modular structure permits the easy introduction of new models and materials properties formulations.

ENIGMA-B 7.3.0 has been extensively validated against data from a large number of experimental and commercial irradiation programmes. The validation database for standard $UO_2$ fuel consists of some 414 rod irradiations from 33 different programmes, including OECD Halden experiments, Riso Fission Gas Projects, Studsvik Ramp Tests, NFIR projects, the High Burnup Effects Programme, and many commercial reactor irradiation programmes.

## 3.2. Thermal-Hydraulics Design Criteria

### 3.2.1. Compatibility of Pressure Drop

The pressure drops across the fixed and follower NOVA E-3 fuel assemblies must be compatible with the existing (reference) fuel at nominal reactor operating conditions.

The pressure drop tests (see 2.1.1) indicated that this criterion was satisfied.

Confirmatory tests were carried out on the LTA fuel in the reactor ponds at Loviisa at room temperature and low mass flow rate. These tests confirmed that the pressure drop over the fixed fuel assembly was almost identical to that of existing (reference) fuel.

### 3.2.2. Departure from Nucleate Boiling

There must be at least a 95% probability, at 95% confidence levels, that Departure from Nucleate Boiling (DNB) will not occur on any fuel rod during normal operation or Anticipated Operational Occurrences (AOOs). This is guaranteed by ensuring that the minimum Departure from Nucleate Boiling Ratio (DNBR) value in the DNB limiting AOO is greater than the DNBR design limit.

The thermal-hydraulics tests (see 2.1) have shown that the existing isolated subchannel methodology for DNB analysis can be applied to BNFL fuel conservatively.

The limiting AOO event for the Loviisa reactors has been identified as unintended control assembly withdrawal. With a NOVA E-3 assembly as the limiting assembly, isolated subchannel calculations have shown that there is significant margin to the DNBR design limit in this fault.

### 3.2.3. Bulk Boiling

Bulk boiling must not occur in any flow subchannel or instrumentation tube of the fuel rod bundles in the reactor.

A bulk boiling analysis was performed at limiting Loviisa normal operation conditions in two ways :

(a) An isolated subchannel analysis was performed. The local enthalpy rise in the hot subchannel was calculated as a function of the bundle average enthalpy rise and a hot subchannel enthalpy rise uncertainty factor was applied.

(b) A subchannel analysis of the hot assembly rod bundle was performed using the VIPRE code. The maximum assembly average power, minimum coolant pressure and maximum inlet temperature were modelled together with a bounding beginning of life assemblywise radial power distribution. Uncertainty factors were applied to both the bundle average and hot subchannel enthalpy rises. The Thermal Diffusion Coefficient (TDC) in the turbulent mixing model was obtained from mixing tests (see 2.1.3) performed on test sections representative of BNFL fuel.

In both methods the maximum subchannel temperature at the top of the heated length was compared with the limit on subchannel exit temperature of 325°C. Method (a) was sufficient to satisfy the design criterion. Method (b) was used to demonstrate that a larger margin to the limit exists when inter-channel crossflows and turbulent mixing are taken into account.

### 3.2.4. Other Criteria

Hydrodynamic instability. Examination of the Loviisa pump characteristics showed that the flow excursion type of static instability cannot occur during normal operation or anticipated operational occurrences. The exit fluid qualities that are produced during even the most onerous anticipated operational occurrences are sufficiently low to preclude the density wave type of dynamic instability. Additional evidence that density wave instabilities will not occur is provided by the data from rod bundle DNB tests and from observed instability events.

Assembly Lift Forces. The upward forces acting on the fixed fuel assembly must be insufficient to cause:

a) the assembly to lift off the bottom core plate during normal operation.

b) permanent deformation of the assembly holddown springs during a hot pump overspeed transient.

Calculations of maximum lift forces, based on the results of the pressure drop tests (see 2.1.1), and of the minimum holddown spring forces have shown that both (a) and (b) are precluded.

### 3.3. Fuel Assembly Mechanical Design Criteria

For all operational loading conditions occurring during the design life, the fuel assembly must meet the stress and fatigue criteria of the ASME Code [11], applicable to Class I components for normal operating conditions and AOO. The fuel assembly component structural design criteria were established for the two primary material categories; namely, austenitic steels and Zircaloy-4.

The stress categories and strength theory presented in the ASME Code were used as a general guide. Verification that the fuel assembly can withstand loads encountered during operation was provided by examining that each of the component parts, joints and connections meet the applicable stress, fatigue and load criteria when the operational loads are applied.

Analytical structural analyses have been carried out to ensure that the assemblies can withstand the mechanical loads during: transportation, fuel handling, reactor scram, breakage of the intermediate rod and a Loss of Coolant Accident.

The analysis of the above mechanical loads has been carried out either by hand calculations using standard textbook formulae or by the use of the general purpose finite element code ANSYS [12] and the explicit finite element code DYNA3D [13] which was used to analyse the transient response of structures.

The results of calculations showed that:

(a) The fuel assembly and its components can satisfactorily withstand the maximum allowed shipping (4g axially and 6g laterally) and maximum handling loads at Loviisa.

(b) The reactor trip analysis predicts that the fuel and assembly strains remain elastic.

(c) The intermediate rod breakage analysis does not predict deformations which could prevent the follower fuel assembly being removed from the reactor following the accident event. The analysis enabled the bottom flow plate thickness to be optimised.

### 3.4. Nuclear Design Criteria

The NOVA E-3 assembly design has undergone stringent evaluation and assessment in the area of Nuclear Design. Separate nuclear designs were developed within the same mechanical design with the intention of supplying lead test assemblies to Loviisa and of licensing full reloads at PAKS. The two power plants have very different operating and safety criteria meaning that an identical nuclear design was not possible.

Nuclear design calculations were completed on a single assembly basis for both reactors using CASMO-4. For Loviisa, whole core calculations were performed for equilibrium cycle and cycles incorporating the LTAs using HEXBU-3D. For PAKS, the exploratory calculations were performed for an equilibrium core loaded entirely with BNFL fuel; these calculations were done using the code KARATE.

During the design procedure, the design was enhanced and modified to ensure compatibility with existing fuel and with the strict safety criteria imposed by the licensing authorities as well as the power plants themselves.

The fuel rod burnup was limited to 48 GWd/tU and 56 GWd/tU, the licensed limits of Loviisa and PAKS respectively. For Loviisa, the energy output of the LTAs over their lifetime in the mixed core was matched to the existing Russian design by using a more optimised design with a lower enrichment. Power peaking within the Loviisa core was constrained to be lower than that with the existing fuel. Calculations also showed that safety limits including local linear heat rate and sub-channel enthalpy rise were not exceeded when relevant uncertainties were applied.

Although the coolant temperature coefficient depends on the loading pattern adopted, it was shown to be negative over all states where the reactor is critical in all cycles modelled for both Loviisa and PAKS. Similarly, the shutdown margin at any operating condition for each cycle has been shown to exceed the minimum requirement for both plants (1% in Loviisa and 2% in PAKS). In addition, both Loviisa and PAKS impose limits on the isothermal recriticality temperatures, these criteria were met for both reactors.

The calculations also showed that the design met the design criteria stating that full boration of the coolant would ensure subcriticality (by 1% Loviisa, 2% PAKS) of the reactor with zero inserted control assemblies. Also, the subcriticality was shown to be greater than 5% in both reactors during refuelling operations where one control assembly is inserted. These criteria were met using the boron concentration of the emergency core cooling water for each reactor.

Finally, calculations have shown that with the same enrichment the BNFL fuel is slightly more reactive than the existing Russian fuel. For assemblies with identical energy output there are no implications regarding pond crticality. The PAKS reload fuel design incorporates a number of gadolinia doped rods to ensure this lower reactivity. Calculations have been completed using the HELIOS 2D transport theory code which is capable of modelling storage racks; these show that the reactivity of the BNFL fuel in the ponds is significantly lower than that of the current fuel.

The nuclear design analysis shows that the BNFL NOVA E-3 fuel design is compatible with, and can be loaded alongside the current fuel. Moreover, it confirms that the Loviisa LTAs and any reloads for PAKS meet the required limits imposed by the plant operators and the licensing authority.

### 3.5. Fuel Rod Design Criteria

NOVA E-3 fuel rod thermo-mechanical performance has been evaluated against several design criteria to ensure that fuel failures during normal operation and AOO are highly unlikely, and thus that the risk of radiological release levels are kept very low. These design criteria include limits for such parameters as stress, strain, pressure and temperature levels, and performance against these limits is assessed by modelling fuel behaviour using ENIGMA-B code. The design criteria are listed below:

1. Clad Stress
2. Pellet-Clad Interaction (PCI)
3. Clad Strain
4. Clad Collapse
5. Clad Fatigue
6. Fuel Rod Fretting
7. Clad Oxidation
8. Clad Hydriding
9. Rapid Energy Deposition
10. Fuel Pellet Melting
11. Fuel Rod Internal Pressure
12. Fuel Rod Growth

The bounding power histories used in the analysis include heat ratings and burnups, which go well beyond customer requirements.

The results of calculations to assess the thermo-mechanical performance of both $UO_2$ and gadolinia doped fuel showed that none of the fuel design criteria will be violated by the anticipated duty at either Loviisa or PAKS NPPs. Fuel rod failure during normal operation and AOO transients is therefore highly unlikely.

## 4. SAFETY ANALYSIS

In order to support the loading of BNFL Lead Test Assemblies in the VVER-440s at Loviisa hot rod transient analyses have been carried out by the National Nuclear Corporation (NNC) in the UK for three Postulated Accidents (PAs) at the Loviisa plant.

The first of the three faults considered was the large break Loss of Coolant Accident (LOCA) for which analysis was performed using NNC's NLOCB code. For the reflood phase of this fault the following safety related parameters have been predicted for the hot rod; clad surface temperature, clad oxidation, fuel melting, clad ballooning and burst behaviour. The two other faults considered are pressurised faults; namely, a control assembly ejection fault and a control assembly group withdrawal fault without reactor trip. Both these faults were analysed with the NNC's NICHE code. For these faults the following safety related parameters have been calculated; radial average peak fuel enthalpy, clad surface temperature and clad oxidation.

In each case the boundary conditions for the hot rod were obtained from the results of plant analyses performed by IVO Power Engineering in Finland or KFKI in Hungary. The IVO PE analyses were carried out for a core made up of reference fuel assemblies; the assumption being made that the inclusion of BNFL's Lead Test Assemblies would not affect the course of the plant transient. The KFKI analyses were carried out for a core made up of NOVA E-3 fuel assemblies. A summary of the results of the hot rod analyses is given below against the relevant Safety Criteria.

The hot rod transient analyses carried out for the LOCA fault using whole core plant transients supplied by both IVO PE and KFKI gave maximum clad temperature considerably less than the licensed limit of 1200°C and the maximum clad oxidation was considerably less than the licensed limit of 17%. The maximum clad strain from the dynamic clad ballooning calculation was insufficient to cause contact between adjacent rods, or clad burst. Furthermore it was predicted to cause a reduction in maximum clad temperatures relative to the base case unballooned calculation. It is concluded therefore that a coolable geometry would be maintained during this fault.

The hot rod transient analyses carried out for the control assembly ejection and the control assembly group withdrawal faults, using whole core plant transients provided by IVO PE and KFKI, gave maximum radial average peak fuel enthalpies, maximum clad temperatures and maximum clad oxidation levels below the radial average peak fuel enthalpy limit of 963 J/g $UO_2$, the clad temperature limit of 1200°C and the oxidation limit of 17%.

## 5. CONCLUSIONS

A new VVER-440 fuel assembly has been designed, manufactured and delivered to Loviisa NPP by British Nuclear Fuels plc. The development programme carried out in support of the new design has been presented in this paper.

It has been demonstrated that all of the detailed design criteria relating to BNFL's VVER-440 fuel assembly design are satisfied in normal operation, Anticipated Operational Occurrences and Postulated Accidents at the Loviisa and PAKS NPPs.

## 6. ACKNOWLEDGMENT

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**Nomenclature**

G - local mass velocity, $kg/m^2s$
$q''_{cr}$ - critical heat flux, $MW/m^2$
p - coolant pressure, MPa
x - local quality
$a_1$, $a_2$, $a_3$, $a_4$, $a_5$ and $a_6$ - optimised CHF correlation coefficients.

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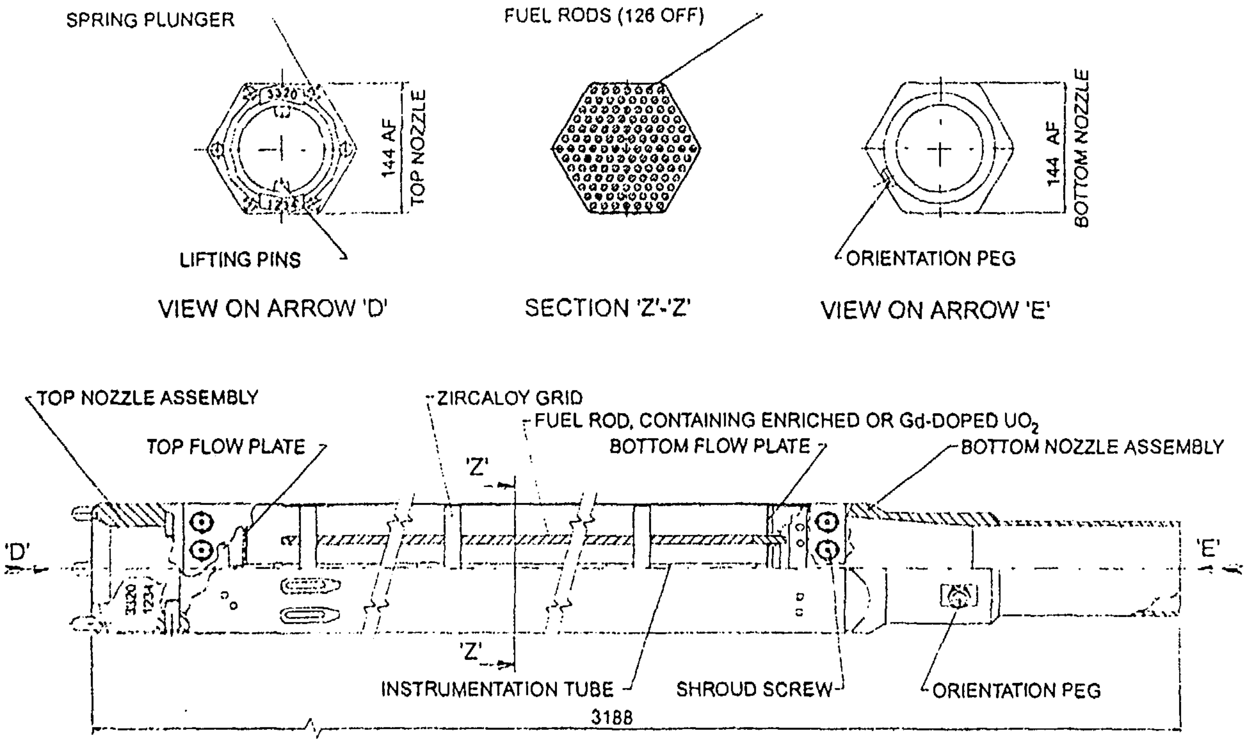
**TABLE 1. BNFL NOVA E-3 Fuel Assembly Design Data**

| | |
|---|---|
| **1. FUEL ASSEMBLY** | |
| Assembly Geometry | Hex |
| No of rods per assembly | 127 |
| Fuelled | 126 |
| Unfuelled | 1 |
| Overall assembly length (mm) | 3188 |
| Overall assembly across flats (mm) | 144 |
| **2. FUEL ROD** | |
| Rod length (mm) | 2520 |
| Rod outside diameter (mm) | 8.90 |
| Pellet length (mm) | 9.15 |
| Pellet outside diameter (mm) | 7.63 |
| Pellet density (% of TD) | 95% |
| Average linear fuel rating (kW/m) | 15 |
| Peak linear fuel rating (kW/m) | 35 |
| Clad material | Zr4 |
| Clad thickness (mm) | 0.55 |
| Maximum clad temperature (°C) | 360 |
| Grid material | Zr-4 |
| Average discharge burnup (MWd/kgU) | >40 |
| Maximum Assembly burnup (MWd/kgU) | >50 |

Figure 1. BNFL NOVA E-3 Fuel Assembly




FIXED FUEL ASSEMBLY

Figure 2. HTRF PWR Flow Loop

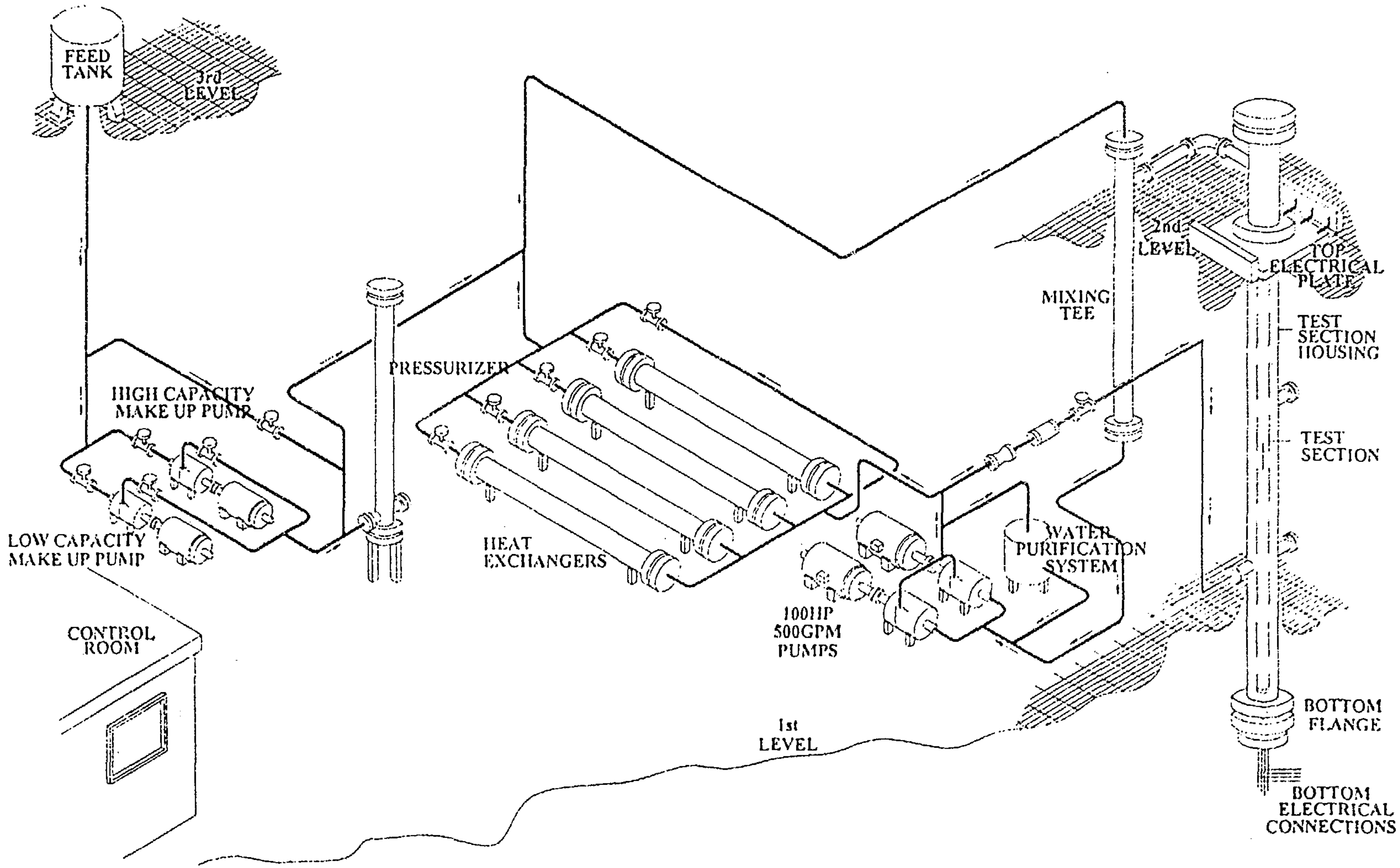


**Figure 3. Pressure Test Instrumentation Arrangement**

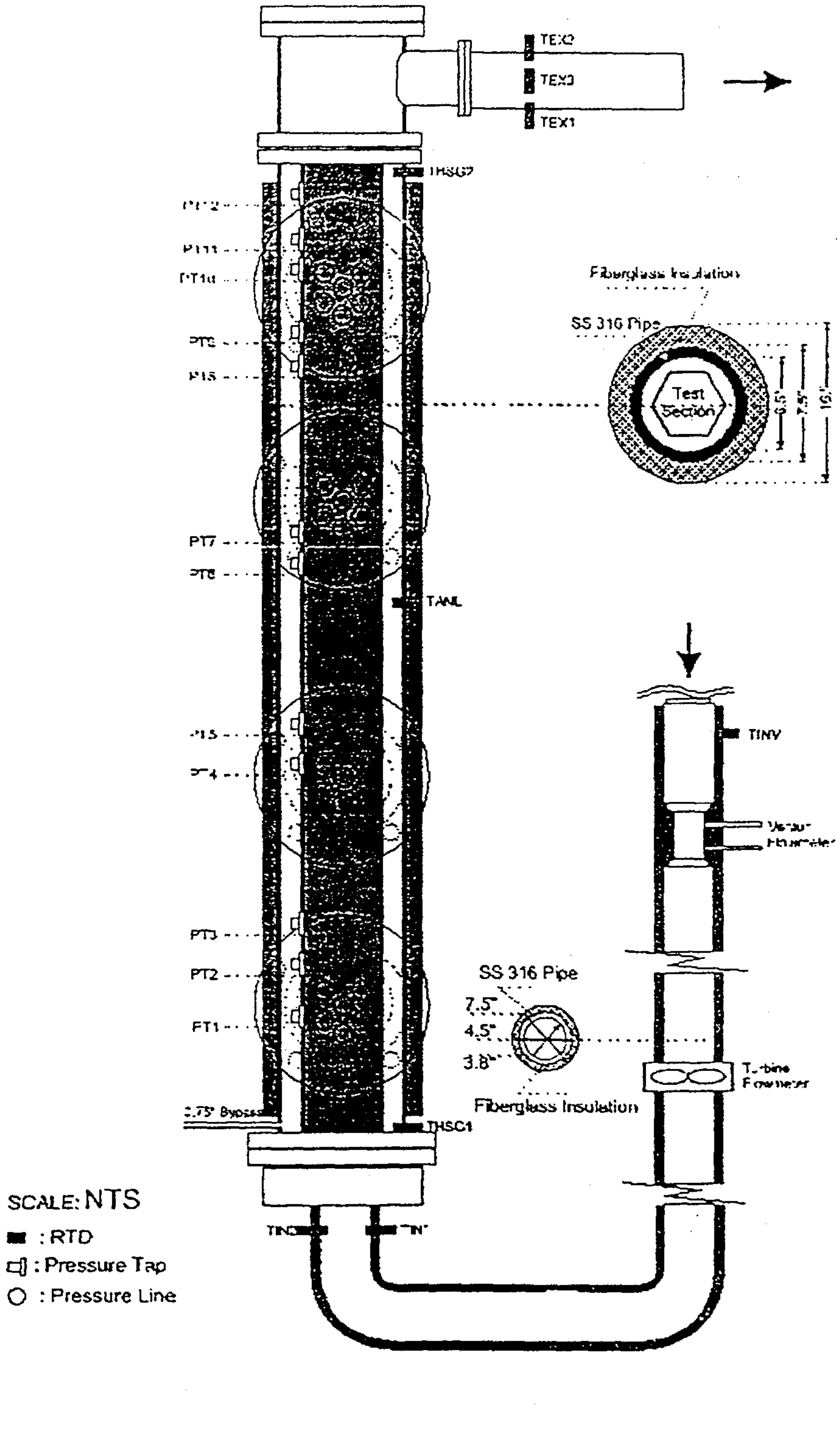
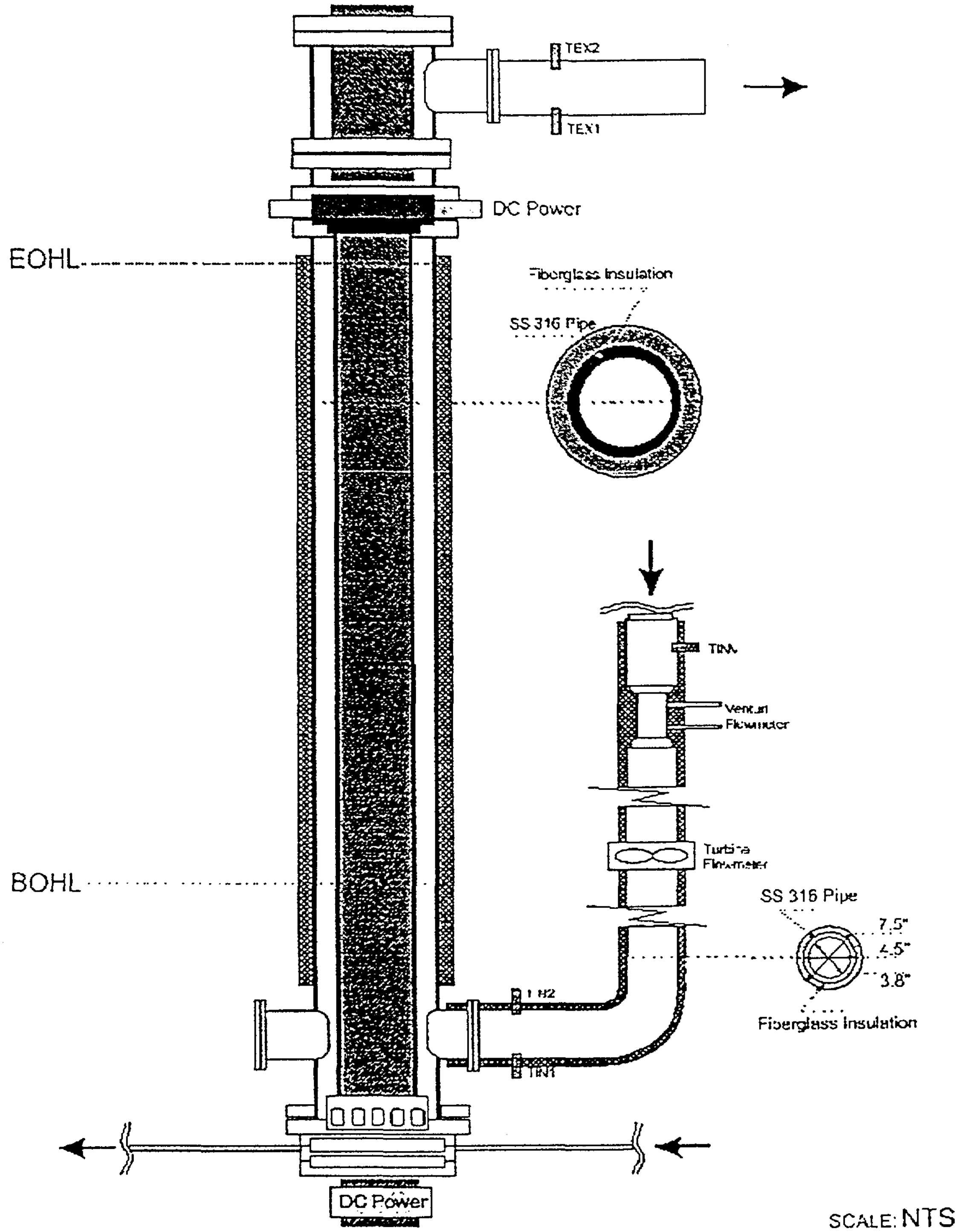


**Figure 4. CHF Instrumentation Arrangement**



## Figure 5. CHF Test Radial Geometry