The Prototype Fast Reactor (PFR)

References:

1. **UKAEA published magazine article on Dounreay, 1987**
2. **PFR training course notes "Introduction to PFR", second edition September 1971**
3. **IAEA-TECDOC-1183, April 1999 "PFR Operating experience"**

PFR was a fast reactor and the name relates to the speed of the neutrons released by nuclear fission (27,000 miles per second!). The reactor has no moderator for slowing down the neutrons and they are less efficient in producing fission in uranium (235) than thermal (slowed down) neutrons, but are more effective in producing new fissile material (plutonium) from natural uranium (238). So while the initial inventory of enriched uranium (or plutonium) is higher for a fast reactor than for a thermal one, the system will produce more fissile material than it consumes (hence the term breeder reactor) and thus eventually utilise most of the energy theoretically available in natural uranium. By contract, a thermal reactor only utilises about 2-3% of this energy and is strongly dependent economically on world uranium prices.

The Prototype Fast Reactor was built to validate and provide operational experience of a large pool-type fast reactor (DFR was a loop-type) and as a test bed for the fuel, components, materials and instrumentation needed for an eventual commercial sized station. It was designed to produce 250MW (electrical) from 600MW (thermal) core power and its design incorporated lessons learnt from early DFR operations. The coolant was sodium rather than DFR's sodium-potassium alloy, as it was cheaper, safer and easier to handle. Coolant flow was upwards rather than downwards to avoid the gas entrainment problems shown at DFR. The fuel was mixed plutonium-uranium oxide in sealed stainless steel pins (DFR had vented enriched uranium alloy) to achieve higher burn-up and to keep the contamination of the coolant to a minimum. The sodium pumps were mechanical centrifugal pumps (electromagnetic pumps at DFR) to obtain higher capacity. Finally, the steam generators were an advanced highly rated tube-in-shell design rather than DFR's low rated, double walled matrix design.

The primary vessel contained 900 tonnes of sodium coolant compared to DFR's 57 tonnes of sodium-potassium alloy. The stainless steel vessel was 12.2m in diameter and 15.2m deep and was encased by a secondary vessel made of carbon steel (for containing leaks). Both were located in an underground concrete lined pit which eased containment and shielding issues.

The nuclear fission heat was extracted from the reactor core by the liquid sodium coolant via three electrically driven (1MW) pumps. The sodium entered the core region at around 400ºC and left the core top at around 560ºC. The sodium then entered six intermediate heat exchangers located in the primary vessel. Sodium in the secondary circuits (75 tonnes in each circuit) flowed through the shell side of each intermediate heat exchanger and transported the heat from the primary sodium to the steam generators. This arrangement was essential to ensure that the core could not be blocked by sodium/water reaction products following a steam generator tube failure, that no active primary sodium was involved in such a reaction and that the primary sodium remained in the biological shield and primary containment.

The three secondary circuits coupled a pair of heat exchangers to three sets of steam generators consisting of an evaporator, a superheater and a reheater. The evaporators were of a forced-circulation type, with each of the three circuits having a steam drum and a boiler circulating pump. This system generated superheated steam from the three circuits and the steam then flowed to a common header to drive the single 300MW turbo-generator and thus produce electricity for the national grid.

The main feed was via a 100% duty steam-driven pump with 10% electric and 10% steam-driven pumps for start up and post-trip use. Appropriate water conditions were provided by a full-flow polishing plant and the feed-heating by sets of low pressure direct contact high pressure tube units. The under slung condenser was cooled by seawater. The seawater pumphouse supplied 480 tonnes of cooling water per minute to the main condenser and the maximum rise in the water temperature was around 10ºC. The local fish, crabs and lobsters thrived on this warm water and when the sea water pumphouse was demolished, very large species were found trapped in the exit chamber.

The reactor core (only 0.91m high by 1.55m in diameter) and its surrounding breeder blanket were made up from an array of hexagonal sub-assemblies, with 325 fuel pins in each. Control was exercised through five boron carbide absorber rods and a further five similar rods were available to shut-down the reactor. A radial breeder blanket surrounded the core and was itself bounded by stainless steel reflector assemblies to improve neutron economy. Outboard of the blanket was a graphite shield which essentially eliminated neutron activation of major removable components such as the primary pumps, valves and the intermediate heat exchangers. Special loops filled with eutectic sodium-potassium alloy (liquid at room temperature) were provided to reject decay heat from the primary coolant via air-cooled heat exchangers, if the steam generators were not available after reactor shut-down.

Fuel could be transferred from the irradiated fuel cave (IFC) to a storage rotor within the primary vessel while the reactor was operating. This rotor system reduced the time required for refuelling operations. The reactor was shutdown only when new fuel was being transferred from rotor to core and/or spent fuel from core to rotor. The spent fuel was generally left in the rotor for one month to allow radioactive decay and to cool, before being transferred to the IFC. Fuel discharged from the rotor after irradiation, was first stored under sodium in the

IFC to cool further and then transferred to the PFR buffer store (water pond), after removing any sodium residues, to await transfer to the reprocessing fuel plant in the fuel cycle area (FCA).

PFR was unique in that it used full commercial sized fuel assemblies, exactly as it would be used in commercial fast reactors. This gave invaluable "hands on" experience of operating such fuel to maximum efficiency, examining the results and reprocessing the spent fuel. The original design target for fuel burn-up was 7.5% and improved design from operating experience eventually produced a world record of 23.2% burn up. The higher the burn up, the less fuel had to be fabricated/reprocessed and the lower the fuel cycle costs.

The fast reactor had three important inherent safety characteristics;

1. The reactor was not pressurised, so there was no danger of a pressure system failure leading to sudden loss of coolant. Even if there was a primary leak, the leak jacket ensured that the sodium still covered the core.
2. If all power to the coolant pumps failed, the sodium could still remove the decay heat by natural convection.
3. The reactor had what was known as a strong negative power coefficient, meaning that as its coolant temperature rose, the power level actually went down. In the event of multiple control systems failure, the reactor would stabilise at about 600ºC, well below the boiling point of sodium and the power level would fall to zero. In effect, the reactor shut itself down.