Status Report - PeLUlt-40 (Indonesia)

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Published Date : Oct 18, 2024 9:54 AM CET

1. Overview

- ☐ **In Operation**
- ☐ **Under Construction**
- ☐ **Described in**

Link to PRIS	--

- ☑ **New Concept**

Number of Units	--
Year	2028

Reference Plant

Name of Site	--
Link to PRIS	--
Net Power Output (MWe)	11.6
Gross Power Output (MWe)	--

1.1 Introduction

PeLUlt stands for *Pembangkit Listrik dan Uap-panas Industri* (Indonesian) that means Nuclear Power Plant for Cogeneration of Electricity and Industrial Heat. In Indonesia the word also means a 'whistle' which represent the spirit and motivation to start the nuclear power plant era in Indonesia. Its initial name was *Reaktor Daya Eksperimental* (RDE). RDE was one of the national programmes to support the National Medium Term Development Plant (RPJMN) for 2015-2019. The main goal of the RDE development Programme is to build the national capability to be a nuclear reactor technology developer by mastering the design, construction project management, commissioning and operation of a nuclear power reactor. Furthermore, the nuclear reactor type selected for the programme should become the prototype design to be scaled up and commercialized to contribute in enhancing the national energy supply security. The Pebble Bed Reactor (PBR) type of High Temperature Gas-cooled Reactor (HTGR) was selected as the technology for the RDE Programme. A sound safety feature and flexible applications of the PBR technology are among the reasons of this decision. PBR has a high safety level shown by a small radioactive release to the environment on any probable accident.

As part of licensing procedure, BRIN (previously BATAN) already received the RDE Site Licensing from National Nuclear Regulatory Body (BAPETEN) in January 2017. Design Approval phase was already started in 2018 and 2019, in which BATAN needs to submit the detail design and safety analysis report of RDE to BAPETEN. However, due to change in main policy of nuclear reactor development, the licensing processes was halted. However, the research and development of RDE are still continuing, particularly related to safety analysis and its cogeneration potential.

Currently there are two target application of PeLUlt-40 as a SMR type cogeneration reactor. First to support the de-dieselization target in Indonesia. It is believed that the size of 40MWt can replace many diesel which currently

expensively utilized in many remote area of Indonesia. PLN as national utility company have a bold program for de-dieselizaton. One of PLN subsidiary, PLN Nusantara Power joined the PeLUIt-40 effort.

Second, as worldwide trend of hydrogen as an important key in energy transition is stronger, cogeneration application for hydrogen production becomes strategic. In particular, a collaboration with an energy state-own company was started to develop a hydrogen production system based on PeLUIt. Based on initial safety analysis, an uprate up to 40 MWt, is possible with generally the same core geometry as the previous RDE design while maintaining its passive safety features.

The general scheme of the PeLUIt system based on HTGR technology is shown in Figure 1. The main components of the nuclear island system are the reactor pressure vessel (RPV) and its internal, coaxial hot gas duct, main steam blower, and the steam generator pressure vessel. The rest of balance-of-plant system is common to other power plants such as a coal power plant. While uprating the power level to 40 MWt, the core and RPV dimension are kept.

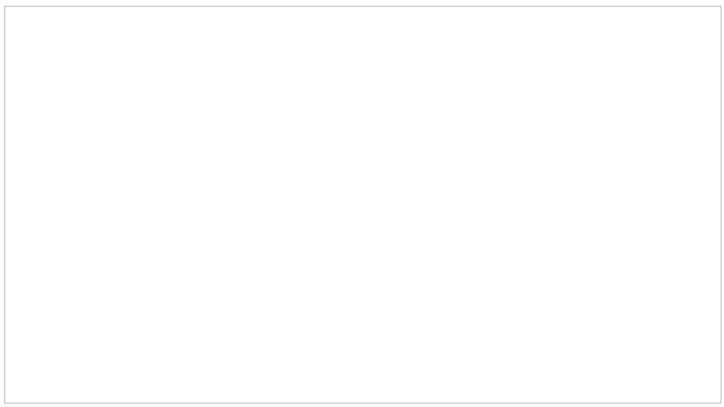


Figure 1. Schematic representation of a HTGR type reactor used in cogeneration mode for the production of hydrogen.

1.2 Development Milestones

2024	Start design approval at Indonesian Regulatory Body (BAPETEN)
2026	Site Licensing
2027	Start Construction and Commissiong Licensing
2028	Start construction of a first full-scale NPP
2031	Initial operation

1.3 Design organization or vendor company

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1.4 Links to designer/vendor website

1.5 Design Description

1.6 Licensing Application Support Documents

In an intensive discussion with Regulatory Body for Design Approval phase which internationally corresponde to Generic Design Review (GDR)

1.7 Reactor Unit in PRIS

2. Category Fields

Table 1: ARIS Category Fields

ARIS Category	Value	Units of Examples
Current/Intended Purpose	Demonstration	e.g. Commercial – Electric/Non-electric, Prototype/FOAK, Demonstration, Experimental
Main Intended Application (once commercial)	Off - grid / Remote, Non-electric (hydrogen production)	e.g. Baseload, Dispatchable, Off-grid/Remote, Mobile/Propulsion, Non-electric (specify)
Reference Location	Inland	e.g. On Coast, Inland, Below-Ground, Floating-Fixed, Marine-Mobile, Submerged-Fixed (Other-specify)
Reference Site Design (reactor units per site)	Single Unit	e.g. Single Unit, Dual Unit, Multiple Unit (# units)
Core Coolant	He	H2O, D2O, He, CO2, Na, Pb, PbBi, Molten Salts, (Other-specify)
Reactor Core Size (1 core)	30 MWth <= Thermal power < 1000 MWth	e.g. Thermal Power < 30 MWth, 30 MWth ≤ Thermal power < 1000 MWth, 1000 MWth ≤ Thermal power ≤ 3000 MWth, Thermal power > 3000 MWth
Reactor Type	GCR	e.g. PWR, BWR, HWR, SCWR, GCR, GFR, SFR, LFR, MSR, ADS
Reactor Size	SMR	e.g. PWR, SMR
Neutron Moderator	Graphite	e.g. H2O, D2O, He, CO2, Na, Pb, PbBi, Molten Salts, (Other-specify)
NSSS Layout	Loop-type	e.g. Loop-type (# loops), Direct-cycle, Semi-integral, Integral, Pool-type
Primary Circulation	Forced (1 pumps)	e.g. Forced (# pumps), Natural
Thermodynamic Cycle	Rankine	e.g. Rankine, Brayton, Combined-Cycle (direct/indirect)
Secondary Side Fluid	H2O	e.g. H2O, He, CO2, Na, Pb, PbBi, Molten Salts, (Other-specify)
Fuel Form	Coated Sphere	e.g. Fuel Assembly/Bundle, Coated Sphere, Plate, Prismatic, Contained Liquid, Liquid Fuel/Coolant
Fuel Lattice Shape	Spherical	e.g. Square, Hexagonal, Triangular, Cylindrical, Spherical, Other, n/a
Rods/Pins per Fuel Assembly/Bundle		e.g. #, n/a

Fuel Material Type	Oxide	e.g. Oxide, Nitride, Carbide, Metal, Molten Salt, (Other-specify)
Design Status	Basic Design	e.g. Conceptual, Detailed, Final (with secure suppliers)
Licensing Status	GDR	e.g. DCR, GDR, PSAR, FSAR, Design Licensed (in Country), Under Construction (# units), In Operation (# units)
Neutron Spectrum	Thermal	e.g. Fast, Thermal

3. Parameter Fields

Table 2: ARIS Parameter Fields

ARIS Parameter	Value	Units of Examples
Plant Infrastructure		
Design Life	40	Years
Lifetime Capacity Factor	90	%, defined as Lifetime MWe-yrs delivered / (MWe capacity * Design Life), incl. outages
Major Planned Outages		# days every # months (specify purpose, including refuelling)
Operation / Maintenance Human Resources		# Staff in Operation / Maintenance Crew during Normal Operation
Reference Site Design	Single	n Units/Modules
Capacity to Electric Grid	11.6	MWe (net to grid)
Non-electric Capacity	3096 kg/day hydrogen with SOEC technology	e.g. MW(th) heat at x °C, m3/day desalinated water, kg/day hydrogen, etc.
In-House Plant Consumption	1.9	MWe
Plant Footprint		m2 (rectangular building envelope)
Site Footprint		m2 (fenced area)
Emergency Planning Zone		km (radius)
Releases during Normal Operation		TBq/yr (Noble Gases / Tritium Gas / Liquids)
Load Following Range and Speed		x – 100%, % per minute
Seismic Design (SSE)	0.26	g (Safe-Shutdown Earthquake)
NSSS Operating Pressure (primary side)	3	MPa(abs), i.e. MPa(g)+0.1, at core outlet
NSSS Operating Pressure (secondary side)	6	MPa(abs), i.e. MPa(g)+0.1, at secondary side outlet
Primary Coolant Inventory (incl. pressurizer)		kg

Nominal Coolant Flow Rate (primary)	17	kg/s
Nominal Coolant Flow Rate (secondary)	14.5	kg/s
Core Inlet Coolant Temperature	250	ºC
Core Outlet Coolant Temperature	700	ºC
Available Tempeature as Process Heat Source	262	ºC
NSSS Largest Component	RPV	e.g. RPV (empty), SG, Core Module (empty/fuelled), etc.
- NSSS Largest Component Length	11100	mm
- NSSS Largest Component Diameter	4200	mm
- NSSS Largest Component Thickness		mm
- NSSS Largest Component Weight	180	tonne
Reactor Vessel Material		e.g. SS304, SS316, SA508, 800H, Hastelloy N
Steam Generator Design	Vertical, Helical	e.g. Vertical/Horizontal, U-Tube/ Straight/Helical, cross/counter flow
Secondary Coolant Inventory		kg
Pressurizer Design		e.g. separate vessel, integral, steam or gas pressurised, etc.
Pressurizer Volume		m3 / m3 (total / liquid)
Containment Type		Dry (single/double), Dry/Wet Well, Inerted, etc.
Containment Volume		m3
Spent Fuel Pool Capacity and Total Volume		Years of full-power operation / m3
Non Electric Application		-
Core Delta Temperature		ºC
Fuel/Core		
Single Core Thermal Power	30	MW(th)
Refuelling Cycle	Continuous	months or “continuous”
Fuel Material	UO2	e.g. UO2, MOX, UF4, UCO
Enrichment (avg./max.)	17%	%
Average Neutron Energy		eV
Fuel Cladding Material	TRISO	e.g. Zr-4, SS, TRISO, E-110, none

Number of Fuel “Units”	27000 sphere	Specify as Assembly, Bundle, Plate, Sphere, or n/a
Weight of one Fuel Unit	0.2	kg
Total Fissile Loading (initial)	0.05/sphere	kg fissile material (specify isotopic and chemical composition)
% of fuel outside core during normal operation		Applicable to online refuelling and molten salt reactors
Fraction of fresh-fuel fissile material used up at discharge		%
Core Discharge Burnup	80	MWd/kgHM (heavy metal, eg U, Pu, Th)
Pin Burnup (max.)		MWd/kgHM
Breeding/Conversion Ratio		Fraction of fissile material bred in-situ over one fuel cycle or at equilibrium core
Reprocessing	None	e.g. None, Batch, Continuous (FP polishing/actinide removal), etc.
Main Reactivity Control	Rods	e.g. Rods, Boron Solution, Fuel Load, Temperature, Flow Rate, Reflectors
Solid Burnable Absorber	B4C	e.g. Gd2O3
Core Volume (active)	5	m3 (used to calculate power density)
Fast Neutron Flux at Core Pressure Boundary		n/(cm²*s)
Max. Fast Neutron Flux		n/(cm²*s)
Core Control Rod Absorber		-
Core Soluble Neutron Absorber		-
Safety Systems		
Number of Safety Trains	Active / Passive	% capacity of each train to fulfil safety function
- reactor shutdown	N/A	- / - / -
- core injection	N/A	- / - / -
- decay heat removal	N/A	- /- / -
- containment isolation and cooling	N/A	- / - / -
- emergency AC supply (e.g. diesels)	N/A	- / - / -
DC Power Capacity (e.g. batteries)		Hours
Events in which Immediate Operator Action is required		e.g. any internal/external initiating events, none

Limiting (shortest) Subsequent Operator Action Time		Hours (that are assumed when following EOPs)
Severe Accident Core Provisions		e.g. no core melt, IVMR, Core Catcher, Core Dump Tank, MCCI
Core Damage Frequency (CDF)		x/reactor-year (based on reference site and location)
Severe Accident Containment Provisions		e.g., H2 ignitors, PARs, filtered venting etc.
Large Release Frequency (LRF)		x/reactor-year (based on reference site and location)
Overall Build Project Costs Estimate or Range (excluding Licensing, based on the Reference Design Site and Location)		
Construction Time (nth of a kind)		Months from first concrete to criticality
Design, Project Mgmt. and Procurement Effort		Person-years (PY) [DP&P]
Construction and Commissioning Effort		PY [C&C]
Material and Equipment Overnight Capital Cost		Million US$(2015) [M&E], if built in USA
Cost Breakdown	%[C&C]/%[M&E]	
- Site Development before first concrete		(e.g. 25/10) (30/40) (20/25) (20/10) (5/15) (----------) (to add up to 100/100)
- Nuclear Island (NSSS)		
- Conventional Island (Turbine and Cooling)		
- Balance of Plant (BOP)		
- Commissioning and First Fuel Loading		
Factory / On-Site split in [C&C] effort		%/% of total [C&C] effort in PY (e.g., 60/40)
Nuclear System		
System Steam Flow Rate		kg/s
System Steam Pressure		MPa
System Steam Temperature		ºC
System Feedwater Flow Rate		kg/s
System Feedwater Temperature		ºC
Reactor Core		
Active Core Height		m
Equivalent Core Diameter		m

Average Linear Heat Rate		kW/m
Average Fuel Power Density		kW/kgU
Average Core Power Density	6	MW/m3
Outer Diameter Of Fuel Rods		mm

4. Plant Layout, Site Environment and Grid Integration

SUMMARY FOR BOOKLET

In general the layout of the building and structures also systems of the PeLUlt-40 can be seen in Figure 2. Detail layout of the PeLUit-40 is under development.

Figure 2. General schematic of building and system arrangement of PeLUlt-40.

Reactor building

Functions, compartment arrangement and equipment layout

The reactor building contains the primary system, important auxiliary and supporting systems and the remote shutdown station. An external shield structure envelopes the inner building structure; it fulfils the requirements for protection of the reactor facility from external events.

The reactor building is subdivided into the following areas:

- The reactor unit
- The services tract
- The erection floor and hall above it (reactor hall)
- The remote shutdown station
- The reactor building annex.

At the center of the modular unit is the primary cavity, consisting of one reactor cavity and one steam generator

cavity. They enclose the pressure vessel unit. Beneath the reactor cavities are the fuel discharge compartment. The primary cavities are surrounded by groups of compartments with various functions.

These compartments contain:

- Entrances to the primary cavities
- Piping and valve stations of the water/steam cycle, the operational
- Component
- Cooling system and the safety-related component cooling system
- Transducers
- Service systems for the primary gas blower
- Auxiliary and supporting systems for the nuclear steam supply system

The services tract consists of two areas. One is between the module units, the other is in front of them.

The services tract contains:

- Portions of the fuel handling Systems common to both modules, i.e. charge stations for new and partially depleted fuel elements, discharge station for spent and partially depleted fuel elements
- Remote shutdown station
- Principle components of gas evacuation systems
- The two closed stairwells
- Elevator.

The remote shutdown station is a walled-off area in the reactor building; it has a separate HVAC system. The remote shutdown station is accessible by way of a secure entrance from the outside during normal operation and on demand. In addition, the services tract is accessible from the remote shutdown station.

The remote shutdown station includes the following rooms and compartments:

- The remote shutdown room with first aid equipment
- A power distribution board compartment
- A battery compartment
- A HVAC system and incoming feeder compartment
- An emergency equipment compartment
- A sanitary facilities room.

The incoming feeder connection for power supply and the hose connections for the secured cooling system (fire hose hydrants) are in the vicinity of the secure entrance from the outside.

The reactor building is served by horizontal and vertical pipe and cable shafts and by way of transport and traffic routes.

The reactor building annex is outside the reactor building shield structure. It houses:

- Pumps and coolers of the operational and secured cooling systems
- Components of the steam generator relief system
- Components of the start-up and shutdown systems.

The following compartments in the reactor building differ in accessibility during reactor operation:

- Primary cavities not accessible
- Equipment compartments accessible under certain conditions
- Service compartments accessible.

The primary cavity contains:

- Pressure vessel unit with internals.

The equipment compartments contain:

- Fuel discharge system
- Fuel handling tube ducts and shafts
- Fuel handling valves
- Main steam inlet branch.

Controlled area boundaries, personnel and equipment access

The Federal German Radiological Protection Ordinance defines the controlled area as that area in which a person spending 40 hours per week could in one calendar year receive a dosage higher than 3/10ths of the limits given in Appendix X, Column 2 of the Radiological Protection Ordinance through external irradiation or incorporation of radioactive materials. The controlled area is specially marked and secured.

The controlled area includes the following plant buildings:

- Reactor building (excluding the remote shutdown station)
- Reactor building annex (middle area only).
- Most of the reactor auxiliary building
- Spent fuel store (set down positions and only when needed for handling fuel shipping casks or inspections, the truck entrance and transport aisle)

The main entrance to that portion of the controlled area comprising the reactor building and the reactor auxiliary building is in the reactor auxiliary building at the + 7.0 m level. The "cold" and "hot" change areas are at the level of the controlled area entrance.

Here, all persons who enter and exit the controlled area are monitored and catered for.

Two stairwells and one service elevator lead to the individual levels of the reactor building.

The reactor auxiliary building also has two stairwells and one elevator.

A truck entrance is provided for the transport of materials and equipment to the reactor auxiliary building. Transport within the reactor auxiliary building is effected through erection openings with the aid of lifting gear or the service elevator.

Materials and equipment are transported into the reactor building through the truck entrance of the reactor auxiliary building, as well. The horizontal transport route and the vertical erection shaft in the reactor building are connected to this truck entrance by a driveway. The reactor building crane can reach all necessary destinations in the reactor building.

4.1 Site Requirements during Construction

Construction and layout of essential structures

The main structures of the power plant are the:

- Reactor building with the reactor building annex
- Reactor auxiliary building
- Switchgear and emergency supply building
- Turbine building.

The service structures include the:

- Spent fuel store
- Operations building
- Central gas supply systems building

- Cooling tower and cooling tower pump structures
- Other auxiliary structures such as the vacuum high-voltage substation
- Demineralized water tanks, pipe bridges, cable ducts, vent stack, etc.

4.2 Site Considerations during Operation
4.3 Grid Integration

5. Technical NSSS/Power Conversion System Designn

SUMMARY FOR BOOKLET

Nuclear steam supply system of the PeLUlt-40 consists basically of:

- The reactor pressure vessel with core, core internals, shutdown systems and systems for feeding and discharging fuel elements
- The gas duct pressure vessel with hot gas duct
- The steam generator with tube bundle and primary gas blower.

The reactor, the heat source of the plant, is located at a higher elevation than the steam generator in this system. Natural circulation in the primary system is restricted in such a way that unacceptable temperatures cannot occur in the metallic portions of the system on loss of cooling.

Under all accident conditions, the reactor pressure vessel, including core internals and shutdown systems and the concrete structure of the reactor cavity can be protected from unacceptable temperatures by the cavity cooler alone.

The pressure vessel unit of the nuclear steam supply system consists of:

- The reactor pressure vessel, including:

1. Nozzles, closure head
2. Fuel discharge tube with forged failed fuel separator block
3. Fuel feed connection tube with valve bank
4. Small ball shutdown unit connection tubes with valve banks
5. Electrical and I&C penetrations.

- The steam generator pressure vessel, consisting of steam generator pressure vessel portion and blower pressure vessel portion, including:

1. Nozzles, cover
2. Main steam and feed water nozzles with connecting nozzles to secondary side systems
3. Electrical and I&C penetrations.

- The gas duct pressure vessel.

Because of the arrangement of the gas flow channels, hot gas does not come into contact with the pressure vessel unit at any point. It is thus possible to use pressure vessel technology proven in light water reactors.

The same applies to the feedwater nozzle. Only the main steam nozzle is exposed to high temperatures and is thus constructed in accordance with the technology used in conventional power plants together with extra requirements specific to high temperature reactors.

The pressure vessel unit is equipped with manways at suitable locations.

The primary gas envelope consists of:

- Pressure vessel unit, excluding components of the main steam and feedwater nozzles which carry only secondary coolant
- Steam generator tube bundle with tube-sheets on the main steam and feedwater side
- Primary-side connecting lines, including the valves actuated by the reactor protection system.

As a rule, joints in the pressure vessel unit and all adjoining piping systems are made with leak tight welds.

Flanged connections in the pressure vessel unit are equipped with welded-lip seals or metal ring gaskets. The flanged connections for the reactor pressure vessel closure head and the blower cover are sealed with double metal ring gaskets. Leak tests are conducted after every assembly to ensure that the system is leak tight.

During power operation, the air in the primary cavity is monitored for helium respectively steam so as to detect any possible leaks at the pressure vessel unit.

In-service pressure test with gas takes place in loaded core in every 8 years.

The primary pressure envelope of PeLUlt-40 consists of the reactor pressure vessel (RPV), the steam generator pressure vessel (SGPV) and the hot gas duct pressure vessel (HGDPV), which are housed in a concrete shielding cavity. The material for the RPV is selected based on ASME Section III. The RPV consists of vessel portion, closure head and nozzles. The RPV internals including the ceramic internal and metallic internal, also the control rod and control rod drive mechanism. The metallic internal include the core barrel with guides and supports, lower structure with bottom plate and the top thermal shield. The ceramic internal include all the bottom, side, and top reflector also the outer carbon brick layer.

5.1 Primary Circuit

Reactor Pressure Vessel

The reactor pressure vessel (RPV) consists of forgings welded together and is closed by a reactor pressure vessel closure head which consists of two welded forgings. The closure head and the body of the vessel are bolted together. The reactor pressure vessel bottom head is penetrated in the middle by a large nozzle for fuel discharge. Suspended from this is the fuel element discharge tube with the forged failed fuel separator block. In addition, there are nozzles for fuel element supply and for the small ball shutdown system. The shell course above the bottom head (support lug course) is reinforced. It includes the gas duct pressure vessel nozzle. The three RPV support lugs are fixed to the outside. Above the support lug course is a support ring for internals with a sealing surface for the core barrel. To this cylindrical wrapper plates are connected. The RPV horizontal supports, the stops for the core barrel guide pads, and various nozzles are arranged in a reinforced course in the upper portion of the RPV. Only that portion of the RPV beneath the pebble bed is insulated on the outside. The insulation can be removed for in-service inspections. The main components of the RPV and its internals, including the reactor core can be seen in Figure 3.



Figure 3. Main components of the RPV and its internals.

Gas duct pressure vessel with hot gas duct

The gas duct pressure vessel and the coaxial hot gas duct serve to convey the primary coolant between the reactor pressure vessel and the steam generator pressure vessel.

Hot gas flows from the reactor outlet to the steam generator (SG) in the hot gas duct; cold gas flows from the gas outlet nozzle of the steam generator pressure vessel to the inlet nozzle of the RPV in the annulus between the gas duct pressure vessel and the hot gas duct.

The vessel consists of a seamless forged ring welded to the nozzle of the RPV and the steam generator pressure vessel. It is insulated on the outside.

Working radially outwards from the inside the straight hot gas duct is constructed as follows:

- Metal pipe to guide the hot gas flow and to protect the insulation material from direct fluid flow loadings
- Stuffed fiber mats with V-Shapes convection barriers
- Metal support pipe acting as load-bearing member and as seal between the hot and cold gas regions.

Bellows expansion joints compensate for displacements caused by thermal expansion of the hot gas duct.

In-service inspections of the hot gas duct are not planned.

Steam Generator

The steam generator pressure vessel consists of the steam generator pressure vessel portion and the blower pressure vessel portion joined to the former by bolting. The vessel is located to the side and is installed lower than the reactor pressure vessel.

The steam generator pressure vessel portion consists of several forgings joined by welding.

The bottom torispherical head has a blanked-off nozzle for use as a manway.

The bottom forged ring with the feedwater nozzle and seismic supports is reinforced. The nozzle is welded into the shell around a thermos-sleeve.

The lower cylindrical shell is forged as a seamless ring.

The reducer ring serves to support the cold gas header of the steam generator shell and as a transition between the different diameters of the lower and upper courses of the SG pressure vessel portion.

The upper shell bearing the nozzle for the gas duct pressure vessel, the main steam nozzle and the support lugs for the sliding support, is reinforced. The support lugs are mounted on the outside. The main steam nozzle is welded into the shell around a thermos-sleeve.

The flanged ring at the top of the steam generator pressure vessel portion receives the bolts for fastening the blower pressure vessel portion.

The blower pressure vessel portion consists of the following 3 forgings:

- A flange ring integral with an internal ring forming the supporting flange for the primary gas blower
- A cylindrical shell with various nozzles and an integrated cover flange
- The blower cover.

Pressure control and pressure relief system

Pressure control system

The operation primary system pressure is controlled by feeding helium through the helium purification system to and from the purified gas store (helium supply and storage system).

Pressure relief system

The pressure relief system protects the primary gas envelope from overpressure and has two train configurations. Each train contains an isolation valve, a blocking valve, a safety valve, and a rupture disc. The response pressures of the two safety valves are staggered in such a way that the train with the smaller cross-section responds first. This train alone is sufficient to handle all design basis accidents. In the postulated event of one safety valve being stuck open after response the train concerned is automatically reclosed by the blocking valve to minimize the escape of primary coolant. The isolation valves upstream of these valves are closed only for maintenance, inspection and repairs, and are interlocked in such a way that the valve of only one train can be closed at a time.

The rupture discs downstream of the valves are provided merely for sealing against possible leaks in the affected train; they allow leakage monitoring in the area between safety valve and rupture disc. Both trains of the pressure relief system have a joint letdown line to the helium purification system before the primary system isolation valve. When required, they blow down into the reactor building (reactor hall).

Pressure equalizing system

The pressure equalizing system protects the core barrel in the reactor pressure vessel from unacceptable differential pressure loadings and assures the integrity of the primary coolant envelope when the reactor pressure vessel is open. To equalize the differential pressure across the core barrel, a connecting line which can be isolated externally is provided as an operational train. The isolation valve is closed only after primary gas pressure has been reduced to ambient pressure before opening the reactor pressure vessel, e.g. for inservice inspection.

To compensate for the differential pressure caused by breaks of large cross-sections (up to DN 65), an internal train is provided which opens an internal connection (inside diameter of approx. 200 mm) between the core barrel and the core barrel-to-RPV gaps containing stagnant helium on response of a rupture disc at a differential pressure of approx. 1 bar.

After rupture disc response and reduction of primary gas pressure to ambient pressure, this internal train can be closed manually with the isolation valve on the core barrel in order to re-isolate it from the atmosphere before the reactor pressure vessel closure head is opened.

Primary system isolation

All primary coolant pipes connected to the pressure vessel unit have a cross-section less than or equal to DN 65, or have suitably reduced cross-sections. As a rule, they are provided with a combination of two series-connected valves of which the one nearest the pressure vessel unit can be operated manually and the second (primary system isolation valve) is actuated by the reactor protection system.

Exceptions to this rule are:

- Lines for failed fuel discharge, which have only one valve actuated by the reactor protection system,
- The external line of the pressure equalizing system, which cannot be isolated in the event of a postulated break,
- Instrument lines, which have two isolation valves not actuated by the reactor protection system,
- Lines with blanked-off flanged ends and manually operated isolation valves.

Steam generator isolation and pressure relief systems

Steam generator isolation system

The steam generator isolation system separates the nuclear steam supply system from the conventional water/steam cycle, which is not subject to any safety requirements in the event of any accident or malfunction in which the reactor is tripped. Backflows in the main steam system are also prevented with one check valve.

Whenever the reactor is tripped, the steam generator is isolated by the reactor protection system which shuts off the main steam and feedwater sides by closing two series-connected isolation valves in each line. The reactor protection system also isolates the start-up and shutdown circuit whenever the reactor is tripped during start-up and shutdown operation.

Steam generator relief system

In the event of a steam generator tube break, the steam generator is quickly emptied in order to limit the amount of water which enters the primary system. The pressure relief valves are closed after relief in order to prevent the primary system from depressurizing through the steam generator tube break.

When the accident condition has been terminated, the steam generator and the primary system connected to it by the break are completely emptied through the helium purification system.

Steam generator relief is performed through the feedwater piping. The steam generator relief system is not subject to any requirements relating to overpressure protection of the steam generator secondary side. This is performed by a separate safety valve connected to the steam side.

The steam generator relief system consists of two parallel relief lines for each steam generator. Two relief valves are arranged in series in each relief train.

This parallel and serial arrangement of the relief valves assures redundancy for reliable opening and closing. The two relief lines discharge into a flash tank. The discharged

Feed water is collected in a tank downstream of the flash tank.

For opening the relief valves are actuated by the reactor protection system. The valves are closed with elasticity at pressure equalizing between secondary and primary side.

The steam generator relief system is designed in such a way that no more than 600 kg of water can enter the primary system in the event of a steam generator tube break with only one of the two pressure relief trains operating.

5.2 Reactor Core and Fuel

The equilibrium core consists of approx. 27,000 spherical fuel elements in a loose pebble bed having a diameter of 1.8 m and an average height of approx. 2 m (higher pebble fuel possibly occurred in the center radius due to the recirculating scheme with single drop in the center of the core) and is cooled by helium.

Mean power density during nominal power operation of the core is limited to 8 MW/m3 and the mean core outlet temperature to 700°C, it is result of the uprating from the previous 2 MW/cm3 of RDE design. In this single-zone core, axial power density distribution must be made sufficiently uniform in the equilibrium core..

Under all operational and accident conditions, residual heat can be removed solely by thermal conduction, thermal radiation and natural convection to the surface coolers outside the reactor pressure vessel. Even without active residual heat removal from the core, the maximum fuel temperature of approx. 1400°C is not exceeded.

The core and its geometry are so designed that the reactor can be shut down merely by the insertion of absorbers in reflector columns.

Because of the core design, the total temperature coefficient is so negative that the inadvertent withdrawal of all reflector rods is controlled solely by primary gas blower trip; again, the allowable fuel temperature of approx. 1400°C is not exceeded.

The effect on reactivity of accident-induced water ingress is less than that of inadvertent withdrawal of all the reflector rods. The core length is such that not damped axial xenon oscillations are ruled out.

The primary helium coolant works at the pressure of 3.0 MPa. Helium coolant enters the reactor in the bottom area inside the pressure vessel with an inlet temperature of 250ºC. Helium coolant flows upward in the side reflector channels to the top reflector and top helium plenum and flow into the pebble bed in a downward flow pattern. Bypass flows are introduced into the fuel discharge tubes to cool the fuel elements there and into the control rod channels for control rods cooling. Helium is heated up in the active reactor core and then is mixed to the average outlet temperature of 700ºC and then flows to the steam generator. The hot helium then transfers its energy to the 243ºC feed water in the steam generator to have a superheated steam of 520ºC at 6 MPa flowing to the turbine to generate a ~13 MWe or ~3 MWe.

Fuel elements are spherical ones. Every fuel element contains 5 g of heavy metal. The equilibrium core has 17% enrichment of U-235. Uranium kernels of ~0.5 mm in diameter is coated by three layers of pyrocarbon and one layer of silicon carbon. Coated fuel particles are dispersed in matrix graphite with 5 cm in diameter. Surrounding the fuel containing graphite matrix is a 5 mm thick graphite layer.

For fuel supply, currently BRIN communicates with INET Tsinghua University on the possibility of supplying the HTR-10 fuel. From the beginning, the RDE and currently PeLUt-40 design is based on HTR-10 fuel due to supply reason as developing and testing a new fuel will take a long time beyond the target of PeLUlt-40.Communication with other potential vendors such as ROSATOM and South Africa has also started as they have the experience in fabricating the TRISO-based pebble fuel.

The representation of the fuel in the PeLUlt-40 is shown in Figure 4.

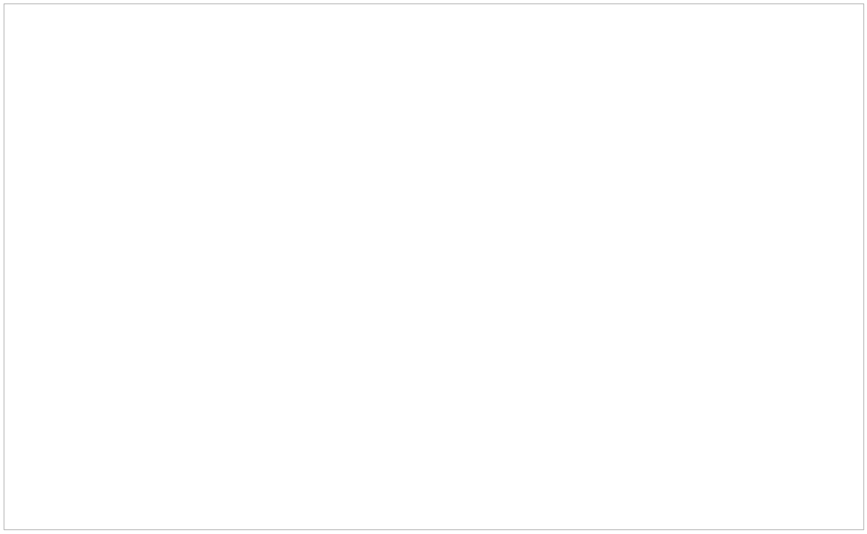


Figure 4. TRISO fuel.

5.3 Fuel Handling

The operation mode of PeLUlt-40 adopts continuous fuel loading and discharging: the fuel elements are pneumatically lifted into the upper part of the reactor, drop into the reactor core using a single fuel loading tube, then move downward across the core and through a discharging tube at the core bottom. The fuels will pass one-by-one through the singulator. The geometry of discharged fuel elements is checked in the fail-fuel separator. Failed fuel with geometrical defects will be separated and diverted into the failed fuel cask, while the good ones will continue to the burn-up measurement facility. Fuel pebbles that already reached the burnup target will be collected in the spent fuel cask while the other will be redirected back into the core. In average, a single pebble fuel will pass the core 5 times to reach the average discharge burnup target of 80 MWd/kg. A Once-Through-Then-Out (OTTO) fuel management scheme is the main operational mode of PeLUlt-40 to have a simpler design which finally improve its techno-economic performance while trying to optimized the burnup to be in the level of 80 MWd/kg.

5.4 Reactor Protection

PeLUlt-40 is equipped with two independent reactivity control or shutdown system, a control rod system and a small ball shutdown system. Two independent and diverse shutdown systems are provided; they are inserted into the side reflector to shut down the reactor.

The shutdown systems are designed and arranged in such ways that, on demand, the absorbers drop into their most effective position solely under the force of gravity.

Only the first shutdown system is controlled by the reactor protection system, while the second shutdown system can be actuated by hand, when needed.

The two shutdown systems are also used for control. In particular, the rod position of the first shutdown system is selected for the equilibrium core in such a way that load changes within a range of 50 to 100% of nominal are possible at any time.

Control rod

Control rods are used for shutdown, fine temperature adjustment and trimming. Each control rod can move in the side reflector columns independently.

The ten reflector rods consist of several elements held together by articulated joints. The absorber material in the form of sintered B4C rings is located between two coaxial tubes in each element. The absorber is cooled inside and outside by a stream of cold gas.

In order to integrate the rod drive mechanisms into the reactor pressure vessel, the absorber length is limited to approx. 2.2 m. A link chain driven by an electric motor through a planetary gearbox serves as the lifting element between the drive mechanism and the absorber proper. On reactor trip, the motor's power supply is cut off, which results in the reflector rod falling under gravity but damped to its lowest position (0.8 m below core center).

Because the rod drive mechanisms are fully integrated into the pressure vessel unit, depressurization accidents via the drive mechanism housing and hence rod ejection due to fluid flow forces can be ruled out.

Each rod is equipped with an analog position indicator which measures the position of the rod over its entire positioning range and with binary position indicators for the upper and lower limit positions.

Small ball shutdown system

The small ball shutdown system is provided for cold and long-term shutdowns. The small ball shutdown elements are stored above the top thermal shield and fall under gravity into reflector columns (slotted holes) by demand. The passive control capability of the reactor is supported by its strong negative reactivity feedback and a low excess reactivity. The 7 small ball shutdown units are distributed as evenly as possible around the perimeter of the core. The shutdown elements are graphite balls with a B4C content of approx. 25 % and a diameter of approx. 5 mm. The small ball shutdown elements are stored above the side reflector on the top thermal shield and fall under gravity into reflector columns on demand.

Each of the 7 storage vessels has a closure which allows the discharge of less than full quantities without causing ball breakage. The power supply to the closure solenoid is cut off on demand, causing the vessel closure to open by gravity and the small ball shutdown elements to fall freely into the slotted holes in the side reflector. Both limit positions of the closure are signalled.

A pneumatic suction system is used to return the shutdown elements from the reflector columns in controlled quantities to the storage vessels. Measuring devices constantly monitor the fill level of the storage vessels, including the "vessel full" and "vessel empty" states.

An interlock ensures that if the reflector rods are actuated by the reactor protection system, the operation of the small ball shutdown system is continued.

The shutdown elements are drawn from vessels integrated into the metallic core support structure, which is filled with balls at all times. The transport fluid is cold gas which is extracted from the primary system beneath the metallic core support structure and which is conveyed to a common carrier gas blower outside the primary system. Sets of control valves for carrier and bypass gas supply serving only 1 small ball shutdown unit are arranged in one valve bank each on the reactor pressure vessel. Each of these valve banks also includes a manual isolation valve and a valve actuated by the reactor protection system (primary system isolation valve).

Servicing of the reflector rod drive mechanisms and components of the small ball shutdown units (e.g. vessel closure) is possible when the reactor is depressurized.

Shutdown by shutdown systems

Equilibrium core

The first shutdown system (reflector rods) renders the reactor subcritical hot from all accident-induced conditions for a sufficient length of time, even if failure of the highest-worth reflector rod is postulated.

The second shutdown system (small ball shutdown system) alone is able to render the reactor subcritical down to the lowest possible operating temperature from all operating states and for any length of time.

Both shutdown systems together render the reactor subcritical for any length of time down to the lowest possible operating temperature from all operating states and accident conditions, even postulating the failure of the highest-worth small ball shutdown unit.

It is ensured by a limitation of the operational level of the reflector rods that in all requested conditions enough shutdown

reactivity is available.

First core and running–in phase

The requirements and functions of the control and shutdown systems in the first core are not fundamentally different from those of the equilibrium core. However, since the temperature coefficient is especially negative during the first months at full load, shutdown margins for the first core are lower than for the equilibrium core. Under accident conditions, if the reflector rods are not used to shutdown or a small ball shutdown unit fails, it would consequently be possible for the reactor to return to criticality at temperatures below 100°C. This condition does not present a problem from the safety standpoint, since the core can only generate an output corresponding to the small power loss. It is ensured by controlling the sub-criticality of the core that this condition will not occur. If the core temperature falls too far, the core can be brought to a higher temperature level, by a heat input caused by primary coolant circulation or by making the core critical for a short time. If the running-in phase has to be interrupted for a long time, the reactor will be made cold subcritical by addition of neutron-absorbing substances.

Shutdown by cutting off primary coolant flow

Another way of shutting down the reactor is to cut off the primary coolant flow; this leads to a slight increase in the mean core temperature and renders the reactor subcritical because of the negative temperature coefficient for reactivity. As a rule, the primary coolant flow is cut off by shutting down the primary gas blower each time the reactor is tripped. In addition, the blower damper is closed by an operational automatic control. A limiting device limits the reactor power to 105% by measuring the speed of the primary circulating blower.

5.5 Secondary Side

The general process flow diagram of PeLUlt-40 including the turbine performance demand is shown in Figure 5. Particular turbine with the capability to have an intermediate steam extraction as needed for low carbon hydrogen production is being isvestigated, several on-market turbine-generator systems are identified.

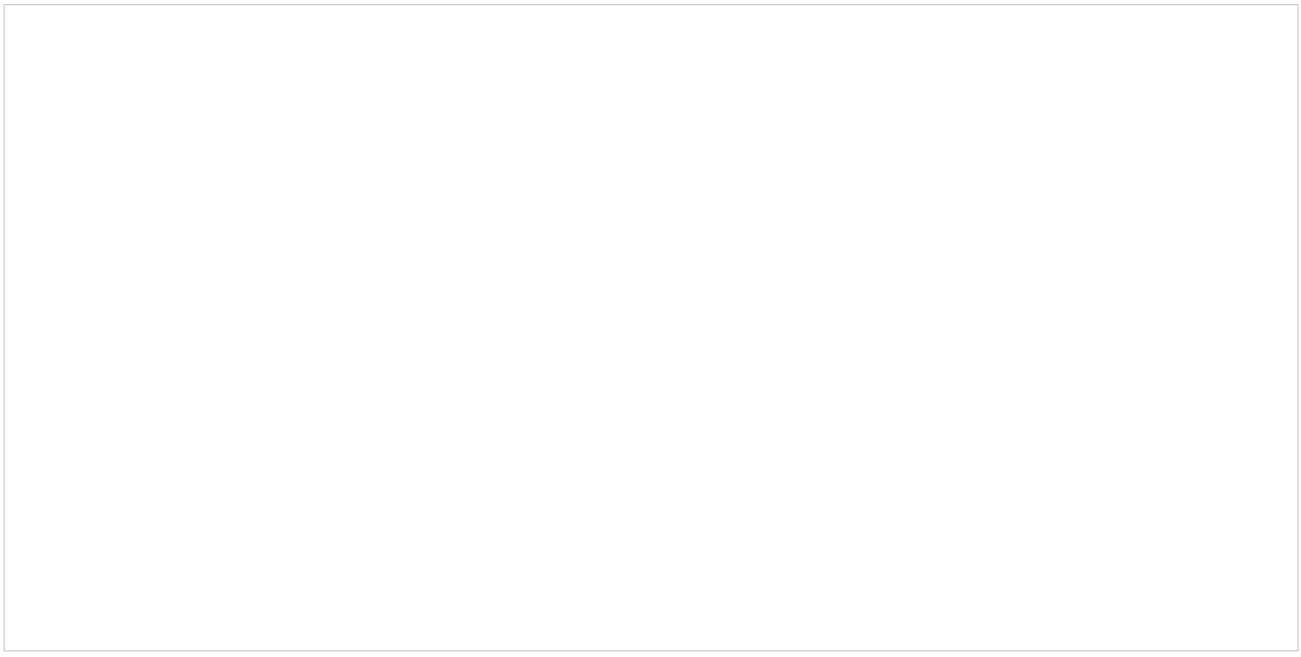


Figure 5. General process flow diagram of PeLUlt-40.

The auxiliary system

In PeLUit-40 the auxiliary system comprises of:

1. Helium Purification System
2. Fuel handling and storage
3. Fire protection equipment

4. Demineralized water supply systems
5. Process auxiliary system
6. HVAC system

Helium is used as the primary coolant for the RDE. A helium purification system provides the necessary degree of helium purity.

In detail, the helium purification system has the following functions:

- Removal of particulate and gaseous contaminants from the primary coolant to maintain design values, in particular for H2O, CO, CO2, N2, H2, CH4
- Removal of tritium
- Removal of other radioactive contaminants from the helium, especially before transfer to the purified gas store (Xe, Kr, Ar)
- Start-up purification of the primary system before initial start-up and after inspections and maintenance
- Purification of newly delivered helium
- Removal of water after a water in-leakage accident

5.6 Containment/Confinement

The confinement envelope of the PeLUlt-40 acts in conjunction with other barriers to the release of radioactive substances to minimize the radiological impact on the environment of accident conditions.

The following features fulfill confinement functions in the PeLUlt-40:

Reactor building

No leak tightness demands are made of the reactor building as part of the confinement envelope or of its penetrations and entrances. The maximum leakage of the reactor building is 50 Vol.-%/d with differential pressure of 2 mbar. A sub-atmospheric pressure system and a pressure relief system are provided in the reactor building merely to minimize the impact on the environment after a postulated break in the primary system. The entrances to the building consist of interlocked doors, which assure that a directional air flow in the building can be maintained at all times.

Since it is possible to relieve building pressure directly into the atmosphere in the event of a depressurization accident, it is unnecessary to design the building for high interior pressures.

The reactor building is designed for the pressures and temperatures which are postulated to occur in the event of a primary coolant pipe break or main feedwater line break and building pressure relief to the environment.

The primary cavities and outside walls of the reactor building are designed against a pressure of 0.3 bar. The shielding effect of the primary cells provides reliable protection of the environment from direct radiation.

Secured sub-atmospheric pressure system

Areas containing primary coolant pipes are monitored for helium leakage. If specified limits are exceeded, the affected area of the building is switched from the unfiltered air exhaust system to the secured sub-atmospheric pressure system. The secured sub-atmospheric pressure system is designed for a volumetric flow rate of 1.25 m3/s and receives emergency power back-up. The system is equipped with a HEPA filter and an activated-carbon absorber. The design of the secured sub-atmospheric pressure system for postulated external events is not envisaged.

Building pressure relief system, HVAC systems isolation

The arrangement of the compartments in the reactor building and the HVAC systems assure that air flows from rooms with low airborne activity levels into rooms with higher airborne activity levels in order to limit any spread of radioactive

materials in the building under normal operating conditions. As far as possible, the secured sub-atmospheric pressure system is used to minimize radioactive releases in the event of accidents.

Relief ports fitted with rupture discs or dampers which connect the reactor compartments with the reactor hall are provided to control depressurization accidents (DN 65).

Building pressure itself is relieved into the environment via the vent stack. Each port is equipped with:

- A pressure relief damper which opens at a response pressure of approx. 0.1 bar and closes automatically after pressure equalization
- An isolation damper which is normally open and can be closed manually after a depressurization accident in case the pressure relief damper sticks open.

Sub-atmospheric pressure can thus be restored after pressure relief. A pressure relief system, charging to the vent stack, is also provided for the equipment compartments (helium tract) in the reactor auxiliary building.

Because the fuel elements themselves provide reliable retention of radioactive fission products and because helium is used as the primary coolant, no special demands are made of the confinement envelope since compliance with the limits as stipulated in the governmental regulations (in Germany: Article 28, Paragraph 3 of the Federal German Radiological Protection Ordinance) is assured even without additional fission product retention facilities.

In principle, it is possible to discharge the primary coolant unfiltered directly into the environment in the event of accident-induced depressurization of the primary system.

Nevertheless, it is intended to minimize the radiological impact on the environment of a depressurization accident by filtering the escaping primary coolant in the secured sub-atmospheric pressure system before discharge from the vent stack. The design basis for the system is a primary system break with a diameter less than or equal to 10 mm. For larger postulated breaks which lead to a significant pressure build-up in the reactor building, building pressure is relieved by direct discharge of primary coolant into the environment. After depressurization, the pressure relief ports are closed; further discharges via the secured sub-atmospheric pressure system can be filtered.

5.7 Electrical, I&C and Human Interface

Core instrumentation

The neutron flux instrumentation is an ex-core instrumentation system in which probes in probe guide tubes are distributed radially and axially in the cement structure of the reactor cell in such a way as to:

- Monitor integral core power
- Detect the general structure of axial power distribution, and
- Detect azimuthal asymmetry

The ex-core instrumentation system consists of the following instrument channels:

- One instrument channel group for the start-up range (source range)
- One instrument channel group for the intermediate range
- One instrument channel group for the power range
- One instrument channel for the intermediate range with indicator at the remote shutdown station.

Altogether, the detectors monitor core power from the subcritical cold state up to 200% of nominal power, and also macroscopic power distribution. For this purpose, a neutron flux density range of about 11 decades has to be detected at the measurement point.

The source range and the logarithmic intermediate range together cover the total range up to double nominal power. The source range covers the lower 5 to 6 decades, the logarithmic intermediate range approximately the upper 6 decades.

The linear power range detects the upper two decades of the neutron flux density. The constant values derived from the power range make it possible to detect incorrect positioning of the reflector rods or the inadvertent insertion of small ball

shutdown elements, as well as to determine core power.

In order to assure an adequate background for the source range detectors, a neutron source is provided for reactor start-up. It is located in a column branching off a reflector rod column in the side reflector in the upper half of the core. No additional secondary source is provided since so many photo-neutrons are produced by fission product decay in the reactor core after a few months at full load that the plant can be safely started even without an external neutron source.

In order to monitor the criticality for loading of the first core, an additional instrumentation is provided.

Thermocouples are installed at several locations in the ceramic internals of the core area as part of the operational instrumentation. Measurements from them serve to validate the analyses of temperature distributions in this area in various operating states and to quantify the relationships between important operating variables in the first years of plant operation. Because of the limited period of use, non-replaceable thermocouples are installed.

Accident power supply

The accident power generation and distribution system supplies energy to those electric loads which are necessary to maintain safe conditions in the event of loss of auxiliary power. The RDE is so designed, that the limitations of the design range are also respected during long-term loss of auxiliary power and emergency power supply. If the power supply cannot be restored from the auxiliary power supply or from the accident power supply before the expiry of a time period of 72 hours, it will be prevented by external feeding in the cavity cooler through fire brigade connection so that the design temperature of the reactor pressure vessel is limited.

Accident power network in the switchgear building

A two-train network powered by 2x100%-duty accident power supply corresponding to the 2x100%-duty redundancy requirements of the process loads is installed in the earthquake-proof switchgear building.

The accident power network consists of the following sub-networks:

- Accident three-phase AC network for loads which can accept a power cut after loss of auxiliary power until the diesel engines have started.
- A DC network for loads which must remain in operation without interruption or which must be switched on during the interruption before the diesel engines have started in the event of loss of auxiliary power.

The accident power supply to these networks is assured by the following components:

- Diesel generator sets designed for each train's maximum emergency power requirement. They are started either by an operational automatic controller or manually.
- Batteries designed to supply power immediately to the connected loads for at least 2 hours after loss of auxiliary power.

Accident power network in the reactor building

An additional single-train accident power network is installed in the reactor building for those loads which must remain operable for monitoring of the plant from the remote shutdown station on loss of the two-train accident power supply in the switchgear building, e.g. by breakdown of the station service supply and non-availability of the power diesels (longer than 2 hours) due to events such as aircraft crash or blast wave. Consequently, this additional emergency power network is designed for aircraft crash and blast waves.

The single-train accident power network in the reactor building is served by the emergency power supply in the switchgear building when the latter is available.

On loss of the accident power supply in the switchgear building, the single-train network is powered by batteries.

The batteries for the remote shutdown station in the reactor building are designed for a 72-hour discharge period. A facility for connection of cables to external sources is provided.

Reactor protection system

The reactor protection system monitors and processes essential process variables in order to detect malfunctions and accident conditions, and automatically initiates protective actions. The following process variables are monitored and processed

- Neutron flux
- Hot gas temperature
- Cold gas temperature
- Moisture in the primary system
- Pressure in the primary system
- Pressure in the secondary system
- Mass flow in the primary system
- Feed water mass flow
- Earthquake acceleration.

All of the initiation criteria derived from these variables are used to actuate jointly the following protective actions regardless of the initiating event:

- Reflector rod drop
- Primary gas blower trip
- Steam generator isolation.

In addition, the primary system is isolated in the event of a pressure loss in the primary system, and steam generator relief is initiated in the event of water in-leakage into the primary system. Since only one process variable (moisture) is available for initiating steam generator relief, the moisture instrumentation is designed of accordingly high-grade, e.g. in the sense of the German KTA 3501.

Limitations and interlocks for the shutdown systems are decided.

The reactor protection system is installed in the switchgear building and is earthquake-proof. Additionally it is designed against air plane crash and explosion blast wave in the reactor building. So it is assumed that on demand, the following reactor protection actions will be initiated or not prevented by an air plane crash or blast wave or by any consequential damage which these might cause:

- Reflector rod drop
- Primary gas blower trip
- Steam generator isolation
- Primary system isolation
- Steam generator relief.

The cabinet groups in each redundancy are supplied with power from the two 220 V DC system boards of the emergency power system; the feeders are diode-decoupled and incorporate DC/DC converters.

At loss of power supply of the reactor protection system the reactor protection actions are actuated. The reactor protection actions remain actuated with design-measures (storage of reactor protection output signals, low voltage control of cabinet power supply) by voltage recovery. The power plant is in a controlled safe condition; therefore no further monitoring with the reactor protection system is necessary. The reactor protection actions are removed manually, if the power plant is again in normal operating conditions.

The reactor protection system, the control room, and the computer are air-conditioned by a two-train air conditioning system with accident power back-up. In normal operation and in accident power mode, the coolers are supplied by the secured chilled water system in the switchgear supply building. Chilled water is supplied by two air-cooled water chillers.

Control toom

The layout of the control room is under development, however several description can be provided.

Functions of the central control room

The control room provides central control for operation of the PeLUlt-40. It houses the operating and indicating equipment for managing and monitoring of the unit; manual control, set point adjusting and monitoring of the nuclear steam supply systems, the reactor auxiliary systems, the water/steam cycle, the turbine, the generator and the auxiliary power supply systems can be performed from there as required. Additional display equipment for the fire alarm system, interior intrusion detection system and, for example, for monitoring of elevators and entrances are also located in the control room complex. The operating equipment required for communication systems is also located in the central control room. Furthermore, measurement and recording equipment used during and after accidents and beyond-design basis event sequences are also located in the control room complex to:

- Provide adequate information on the condition of the plant to enable the required protective actions to be taken to protect personnel and equipment
- Indicate and record trends
- Permit assessment of any impact on the environment.

The central control room is equipped in consideration of the fact that the PeLUlt-40 is provided with automated instrumentation and control equipment. The equipment described below is provided to satisfy safety and availability requirements. It also relieves the personnel of the routine activities of normal running of the plant, leaving the operators able to intervene in the process by means of the instrumentation and control equipment provided in the central control room, remote shutdown station (only for actuating small ball shutdown system) and local control stations.

The protection systems, i.e. systems which override manual actions and open and closed loop controls, ensure the safety of personnel, of the environment and of the entire plant complex. They encompass the reactor and equipment unit protection systems.

- The reactor protective system detects safety-related plant state variables, processes them and initiates the appropriate actions (reactor scram, primary gas blower trip, steam generator depressurization, isolation of secondary system, isolation of primary system). Reactor protection signals and the actions they automatically trigger override manual actions taken in the central control room and the remote shutdown station and signals from the operational instrumentation and control systems.
- The equipment unit protection system contains the high-grade protection circuits for important equipment units and the protective interlocks for all equipment units, thus protecting these active components from damage or malfunction caused by unacceptable operating conditions or operator errors.

Ergonomic principles are observed in design. This provides operating personnel with a free choice of position (sitting or standing) for the performance of their duties without undue static or dynamic strain.

A further consideration is the functional layout of the equipment in the control room according to the tasks performed:

- The master control section is used for output-related process control, start-up and shutdown of the plant from and to defined process conditions. A serial handling and monitoring system is available.
- The systems control sections is used for running the plant's auxiliary and supporting systems independent of output, and for the preparation for start-up and the performance of periodic tests. Back-up equipment for the nuclear steam supply system and parts of the main systems which are central to availability, are also located in the systems control section. Back-up equipment comprises the control tiles and information equipment (indicators, annunciator panels) and is provided in parallel to the serial video screen input (keyboard) and observation facility. They permit direct intervention at the individual control level and provide information from the individual control level by-passing the automation units.
- The communications area is used for communication between control room personnel and personnel in the plant and with public services.
- The recording area (outside the control room) houses plotting and recording equipment which is not directly involved in the control of the process.

5.8 Unique Technical Design Features (if any)

6. Technology Readiness

SUMMARY FOR BOOKLET

The main equilibrium design and some important severe accident analysis (depressurized loss of forced-cooling) as the enveloped of the safety of PeLUIt-40 were completed.

Verification and code-comparison is highly needed and in progress. Basic design level of engineering development including the process engineering, mechanical and piping, instrumentation and control, electrical, site, civil and architecture is being developed following the international code and standard such as ASME. The design and safety analysis development in progress is part of the targeted approval design which will begin in 2024. Currently the main institution involved in the R&D and design of PeLUIt-40 is BRIN, Bandung Institute of Technology, supported by Technology Development Division of PT. PLN Nusantara Power, Kakiatna Engineering, dan Universitas Pertamina. As an early estimation, the approval design of PeLUIt-40 with BAPETEN will take 2 years.

6.1 Deployed Reactors
6.2 Reactors under Licensing Review
6.3 Reactors in the Design Stage

The main equilibrium design and some important severe accident analysis (depressurized loss of forced-cooling) as the enveloped of the safety of PeLUIt-40 were completed.

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7. Safety Concepts

SUMMARY FOR BOOKLET

7.1 Safety Philosophy and Implementation

PeLUIt-40 is based on the safety philosophy of pebble bed HTGR type which include:

1. **Defence in depth**. The design of the PeLUit-40 should provide defence in depth in which all actions of safety have to set up multiple defenses, so that individual failures have no consequences. In general it include: prevention (and elaborate design, construction and operation to prevent malfunctions), protection (engineering systems to prevent development from malfunctions to accidents), and mitigation (engineering system to mitigate and to limit the consequences of accidents).

2. **Multiple barriers.** As in the pebble bed design, the multiple barriers are: the spherical fuel elements with TRISO, the primary cooling system pressure boundary, the safe confinement.
3. **Design principles** which include:
    - Inferent safety feature of modular pebble bed HTGR
    - Features of instrumentation, control and electrical equipment are as follows: fail safety, redundancy and diversity, to have a high reliability.
    - Stringent and overall quality assurance for safety important components
    - Periodical test of safety related components.

General safety requirements

- Means should be provided to safely shut down the reactor and to maintain it in the safe shutdown condition during and after appropriate operational states and accident conditions.
- Means should be provided to remove residual heat from the core after reactor shutdown, and during and after appropriate operational states and accident conditions.
- Means should be provided to reduce the potentiality for the release of radioactive materials and to ensure that any release is within prescribed limits during and after operational states and within acceptable limits during and after accident conditions.

Safety functions:

- To realize shutting down the reactor safely and maintaining it in the safe shutdown of operational states and accident conditions.
- To prevent anticipated operational occurrences from leading to accident conditions and to shut down the reactor to mitigate the consequences of accident conditions;
- To remove sensible heat and residual heat from the core during and after all operational states and accident conditions;
- To ensure necessary services (e.g. electric, pneumatic etc.) as a support function for a safety system;
- To maintain the integrity of the reactor coolant pressure boundary;
- To keep the radiation exposure of the public and site personnel within acceptable limits during and after accident conditions, in which radioactive materials are released from sources;
- To limit the discharge or release of radioactive waste and airborne radioactive material below prescribed limits during all operational states;
- To maintain control of environmental conditions within the HTR-10 for the operation of safety systems and for personnel habitability necessary to allow performance of operation that are important to safety;
- To maintain control of radioactive releases from irradiated fuel transported or stored outside the reactor coolant system, but within the site, during all operational states;
- To remove decay heat from irradiated fuel stored outside the reactor system, but within the site;
- To maintain sufficient subcriticality of fuel stored outside the reactor coolant system, but within the site;
- To prevent the failure or to limit the consequences of failure of a component or structure whose failure would cause the impairment of a safety function;
- To perform radioactive waste management.

General safety purposes on barriers against release of radioactivity

The PeLUit-40 uses fuel elements in which the uranium fuel is distributed among many small fuel particles each coated with two high-density layers of pyrocarbon and one layer of silicon carbide and embedded in a carbon matrix with an unfueled edge zone.

One characteristic safety feature of the PeLUit-40 is that radioactive substances produced during nuclear fission are confined within the fuel particles during all operating and accident conditions in such a way that there can be no significant release of radioactivity from these fuel particles.

This safe confinement of radioactivity is assured by the design criteria of the fuel particle coatings and the inherent upper limit on the maximum possible fuel temperature under accident conditions. The radiological calculation is based on

1400°C, taking into account uncertainties of computing and other margins. For the design and requirements for the fuel the basis is still 1600°C.

The silicon carbide layer, in particular, is demonstrably so dense up to a temperature of approx. 1600°C that no radiologically significant quantities of gaseous or metallic fission products are released from intact particles. For design purposes, however, it is postulated that a small portion of the approx. 2 x 1E8 coated particles in the core have manufacturing, radiation or accident-induced defects.

Design basis is a proportion of up to 4 x 1E-4 defective particles at maximum accident temperature of approx. 1400°C, so that an average of less than one defective particle can be assumed for each fuel element, taking into account the distribution of burnup and fuel temperature in the reactor core (only approx. 1% of fuel elements experience accident temperatures greater than 1250°C).

Some of the radioactive substances released from these defective particles are retained within the fuel element matrix. That portion which is not retained goes into the primary coolant and is distributed in the primary system. The gas-borne activity in the primary system decreases as a result of radioactive decay, separation in the helium purification system and deposition on the surfaces of the primary system.

The primary gas envelope thus forms the next barrier against the release of radioactive substances. The components of the pressure vessel unit are designed in such a way that during the reactor service life (40 years) through-wall cracks can be ruled out. Because of the quality assurance measures taken, not isolable breaks in the connecting piping are highly improbable

In the event of a break, which is nevertheless postulated, it is by and large only the very slight gas-borne activity in the primary coolant and a portion of the activity deposited on the surfaces of the primary system which could be released into the reactor building. Therefore, no leak-tightness requirements are placed on the reactor building of the PeLUlt-40 to comply with accident dose limits of the Indonesian Regulations (In Germany: as per Article 28 Paragraph 3 of the Federal German Radiological Protection Ordinance) mainly because of the high retention capacity of the fuel particles. Merely to minimize the impact on the environment of a postulated primary system break, the reactor building is provided with a sub-atmospheric pressure system and a pressure relief system.

Technical safety purpose on inherent safety

The engineering configuration and nuclear design of the RDE is such that, even in the event of postulated failure of all active shutdown and residual heat removal systems, the fuel temperature stabilizes at approx. 1400°C.

This is possible because a temperature differential of approx. 750 K is maintained between the maximum allowable fuel temperature and the maximum operating temperature of the fuel elements in the RDE. On account of the negative temperature coefficient for reactivity this temperature differential assures that the reactor core shuts itself down before the temperature limit of 1400°C mentioned above is reached, even in the presence of accident-induced excess reactivity.

In addition, residual heat can be dissipated from the reactor core to surrounding components and structures solely through physical processes (thermal conduction, radiation, convection) because of the selection of a low mean power density in the reactor core, the selection of a suitable geometric design of the reactor core and the surrounding core internals, and through the use of suitable materials.

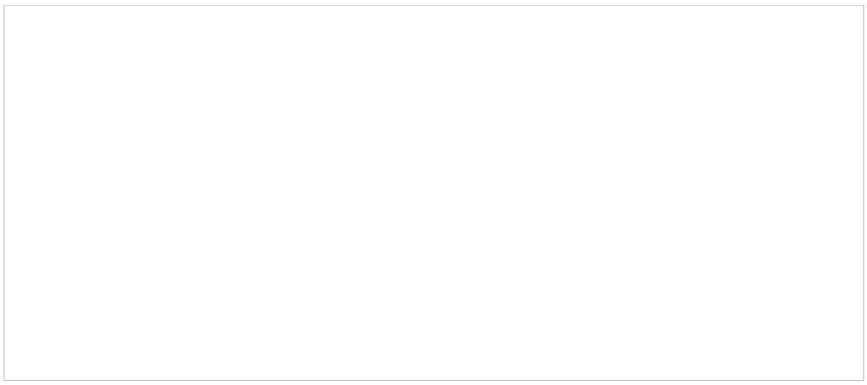
Active residual heat removal systems which limit the loadings on these components and structures can fail for several hours without the allowable limits being exceeded.

No safety-related requirements are placed on the water/steam cycle and the start-up and shutdown systems. These systems are designed and operated as purely conventional plant items.

7.2 Transient/Accident Behaviour

The operation limit of PeLUit-40 are given in Table 1.

TABLE 1. Operation limit of PeLUit-40.



Initiation criteria and protective actions

The initiation criteria provided in the reactor protection system for RDE and the consequent protective actions are summarized in accordance with the accident categories to be considered:

- Reactivity accidents
- Loss of flow events in the primary and secondary system
- Primary system depressurization
- Steam generator tube breaks
- Earthquake

To be able to identify accidents clearly and distinguish them from operational excursions, the initiation criteria are formed by the process variables, their rate of change or the logical combination of several process variables or their rate of change. The logical combinations of process variables are given by the actuation signals: thermally corrected neutron flux and mass flow ratio (primary side and secondary side).

The protective actions thus initiated are:

- Reflector rod drop
- Primary gas blower trip
- Isolation of the steam generator, main steam and feed water lines
- Isolation of primary system
- Steam generator relief

Irrespective of the initiating accident, the first three protective actions in the list (hereinafter referred to as module trip) are initiated in response to all initiating criteria.

Additionally initiated protective actions are:

- Primary system isolation for the primary system depressurization accident category
- Steam generator relief for the tube leakage accident category.

In case of loss of offsite power supply with unavailability of the power diesel and after loss of direct current power supply all five protective actions are initiated automatically without an existing initiation criteria.

Reactivity accidents

This accident category is detected on the basis of initiation criteria derived from the process variables neutron flux or hot gas temperature. Depending on the type of accident (reactivity increase or decrease) the initiating criteria used for module trip are:

- Thermal corrected neutron flux not less than approx. 120 %
- Negative sliding limit for thermally corrected neutron flux not less than approx. 20 %/min

or as an alternative initiating criterion:

- Hot gas temperature not less than approx. 750°C.

Module trip in the event of reactivity accidents during start-up is actuated either in response to the initiating criterion:

- Thermal neutron flux not less than approx. 120 %,

or to both limits of the initiating criterion:

- Intermediate-range neutron flux not less than maximum
- Period not longer than approx. 20 s

The three protective actions leading to module trip transfer the plant to a safe condition and obviate any further safety-related actions.

Loss-of-flow events

This accident category is generally associated with a decrease in primary or secondary system mass flow, the initiating criterion which can be used for such events is

- Mass flow ratio (primary to secondary side) is not less than 1.3 or not greater than 0.75

For secondary-side breaks, module trip can be initiated in response to the initiating criterion:

- Cold gas temperature not less than approximately 280°C
- Negative sliding limit value of main steam pressure higher or equal to 8 bar/min

The mass flow signal is adjusted such that very brief breaks in power supply to the feed water pump or the primary gas blower (e.g. during changeover operations of the feed water pump) do not cause trip. Furthermore the limit value of the mass flow rate is adjusted depending on the hot gas temperature in a way that response of the flow rate signal is prevented during part load operation and during startup and during shutdown operation.

In this accident category protective actions other than the 3 for module trip are not required either.

Depressurization accidents

To mitigate the consequences of accidents in this category, and to trip the module, the reactor protection system also initiates the action:

- Primary system isolation

This isolation prevents further loss of fluid from the primary system through any break downstream of the primary system isolation valves. Leakages between the pressure vessel unit and the primary system isolation valves are improbable on account of the high-grade mechanical design.

The action is initiated in response to the initiating criterion

- Negative sliding limit for primary system pressure not less than approx. 180 mbar/min

The limit setting is chosen such that response does not occur as a result of operational malfunctions, such as failure of pressure control, and during shutdown operation.

In the event of small leakages or in postulated events in which no credit is taken for this initiating criterion, no additional actions are initiated automatically by the reactor protection system; however the module is tripped, but not until the pressure in the primary system has dropped and the initiating criterion

- Mass flow ratio (primary to secondary side) not greater than 0.75 is met.

Ample time is available for the operator in this case to initiate primary system isolation to mitigate the consequences.

Irrespective of the size of the leakage the secured sub atmospheric pressure system is actuated by an operational instrumentation loop.

Steam generator tube leakages

Tube leakages are always connected with the ingress of water or water vapor into the primary system. To limit in-leakages exceeding operationally acceptable levels in the event of leakages, the module is not only tripped in response to the initiation criterion which is specific to these accidents

- Moisture in primary system bigger or equal to 800 vpm

but the following action is also initiated

- Steam generator relief

Since no other process variables capable of providing automatic initiation of this accident-specific steam generator relief action are available for the detection of tube leakages, equipment for this actuation variable is of higher-grade design than for others. By application of valves which close automatically by spring resistance in case of pressure balance of primary and secondary side depressurization of the primary side is prevented. Further measures to mitigate damage (such as starting up the water separator in the helium purification system) are initiated by the operator and not by the protection system.

Postulates and measures for external events

The concept for plant protection against external events, namely

- Earthquake
- Chemical explosion
- Aircraft crash

consists of a well-matched combination of equipment-based, structural and organizational protective measures which take account of the characteristic safety features of the PeLUlt-40 plant.

The external events listed above can be characterized as follows:

- The earthquake is a widespread event. Vibrations caused by an earthquake affect all plant equipment and the vicinity of the PeLUlt-40 plant.
- The blast wave caused by an external chemical explosion is a widespread impact. Loadings caused by it are primarily limited to the vicinity of the explosion source.
- Aircraft crash is a local impact. Loadings caused by aircraft crash, the impact of wreckage, fuel fire and fumes affect individual buildings of the PeLUlt-40 plant.

The concept for protection against external events is based on the occurrences described above. It also takes account of the following loadings:

- Lightning
- Wind, storm
- Snow, rain, hail
- High water, low water
- Hazardous gases.

Special administrative, organizational and structural measures are taken to counteract sabotage.

Protection concept

Consequences of external events need not necessarily disrupt plant operation or affect safety-related plant equipment. Any impairment of the plant leads to a disturbance of system behavior or system conditions. Therefore no special signals are needed to actuate equipment required for external events; the existing initiation criteria of the reactor protection system are used for this purpose. In order to protect the environment from a release of radioactive materials, the following are assured in the case of external events:

- Reactor shutdown and long-term sub-criticality
- Residual heat removal
- Limitation of the radioactive release.

Shutdown and long-term subcriticality

Actions initiated when a limit in the reactor protection system is exceeded include reflector rod drop and primary gas blower trip. These actions are assured even if the power supply fails following the external event. This is because the reflector rods drop into the scram position under gravity if the power supply to the motor is cut off, while the primary gas blower is tripped on loss of power. The reactor is then in the "subcritical hot" condition. The purpose of the small ball shutdown system is to compensate the reactivity gain which occurs when the reactor cools down. In order to actuate this system, the power supply to the closure solenoids is cut off by hand in the control room or remote shut-down station causing the vessel closures to open under the force of gravity and the small ball shutdown elements to fall freely into the reflector columns. All equipment in the reactor pressure vessel required for shutdown and long-term subcriticality is designed to withstand external events.

Residual heat removal

Residual heat is removed by the secured cooling system to protect the reactor pressure vessel, its internals, and the supports for the pressure vessel unit and the concrete structures of the primary cavity. The secured cooling system and the secured service water system are designed for earthquake. In addition, the secured cooling system is designed as far as the hose connections (fire brigade connections) for aircraft crash and explosion blast wave. If the cooling loops fail, the above cooling loads can be cooled via the hose connections. The time available before performance of this action becomes necessary is about 72 hours.

Limitation of radioactive release

By far the largest fraction of the radioactive inventory of the overall plant is contained in the fuel elements within the primary gas envelope. The reactor building is structurally protected against external events, the relevant internals, e.g. primary gas envelope, are designed to withstand induced vibrations. After external events, the reactor building is accessible for repair work. Totally encapsulating protective gear is necessary because of the possible presence of airborne activity. The remote shutdown station is directly accessible from outside. In addition, it is equipped with a separate HVAC system which prevents any airborne activity from entering the station.

8. Fuel and Fuel Cycle

SUMMARY FOR BOOKLET

PeLUlt-40 as continuation of RDE still utilizing the HTR-10 fuel design as reference fuel. This option is chosen due to long time needed to develop and test another new fuel design which will be beyond the target time of PeLUlt-40 (or initial RDE target).

For PeLUlt-40 the Once-Through-Then-Out (OTTO) fuel recirculation scheme is the nominal operation mode. The pebble fuels only pass the core once, the collected in the spent fuel cask and will be stored in the spent fuel part in the reactor building up to around 20 years. Generally, the fuel cycle option is open fuel cycle. In parallel, a long term TRISO based fuel manufacturing was also performed.

For the FOAK PeLUlt-40, as part of the demonstration phase, although the planned fuel recirculation is the OTTO cycle, the fuel handling will be provided with the capability to recirculate the fuel to the core. So the fuel handling adopts continuous fuel loading and discharging: the fuel elements are pneumatically lifted into the upper part of the reactor, drop into the reactor core using a single fuel loading tube, then move downward across the core and

through a discharging tube at the core bottom. The fuels will pass one-by-one through the singulator. The geometry of discharged fuel elements is checked in the fail-fuel separator. Failed fuel with geometrical defects will be separated and diverted into the failed fuel cask, while the good ones will continue to the burn-up measurement facility. Fuel pebbles that already reached the burnup target will be collected in the spent fuel cask while the other will be redirected back into the core.

8.1 Fuel Cycle Options

PeLUlt-40 as continuation of RDE still utilizing the HTR-10 fuel design as reference fuel. This option is chosen due to long time needed to develop and test another new fuel design which will be beyond the target time of PeLUlt-40 (or initial RDE target).

For PeLUlt-40 the Once-Through-Then-Out (OTTO) fuel recirculation scheme is the nominal operation mode. The pebble fuels only pass the core once, the collected in the spent fuel cask and will be stored in the spent fuel part in the reactor building up to around 20 years. Generally, the fuel cycle option is open fuel cycle. In parallel, a long term TRISO based fuel manufacturing was also performed. For the FOAK PeLUlt-40, as part of the demonstration phase, although the planned fuel recirculation is the OTTO cycle, the fuel handling will be provided with the capability to recirculate the fuel to the core.

Fuel handling

Fuel elements are constantly added to and removed from the core during operation. The fuel elements are forwarded by gravity and pneumatically. Primary coolant at cold gas temperature serves as the carrier gas; in special situations, air or nitrogen is used. A charge station for new or partially depleted fuel elements and the same systems for storing spent or partially depleted elements belong to the fuel handling.

Feed and discharge system:

- The fuel elements are forwarded to the core via the central fuel element feed tube, which can be extended when the core is partially filled. Braking gas bled from the primary system in the region of the top reflector absorbs kinetic energy of the forwarded fuel element, thus decelerating it to an acceptable impact velocity.
- The burnup of discharged fuel elements is determined by measuring the characteristic gamma line of the nuclide Cs 137, which is proportional to burnup. If the target burnup has not been reached, the fuel element is returned to the core. Spent fuel elements are sent to the shipping casks (spent fuel storage).
- Fuel element fragments or out-of-shape fuel elements are extracted in a failed fuel separator and conveyed to a failed fuel cask. For availability reasons, two failed fuel separators are provided.
- The feed tube and the failed fuel separator can be removed without allowing air into the reactor.

New fuel storage

New fuel elements are stored in the reactor auxiliary building in double-walled series-produced 200 l hooped drums. To assure that the contents remain subcritical, the gap between the exterior and the interior walls is filled with fine-grained ferro-boron. It is not designed for external events.

Spent fuel storage

Spent fuel elements are stored in shipping casks which were developed for pressurized water reactor fuel assemblies. The design of the casks fulfils the following requirements:

- Shielding of fuel element radioactivity
- Retention of fission and activation products
- Sub-criticality
- Residual heat removal by natural circulation of air around the cask
- Integrity under impacts such as dropping, collision, fire, earthquake, explosion, etc.

Casks are available with capacities of maximum 45.000 fuel elements. After a cask is filled in the reactor building, it is stored in the spent fuel store to await transport to an ultimate repository. The spent fuel store has a capacity of adequate disposal precautions. The spent fuel store structure is not designed for external events.

8.2 Resource Use Optimization
8.3 Unique Fuel/Fuel Cycle Design Features

9. Safeguard & Physical Security

SUMMARY FOR BOOKLET

9.1 Safeguards
9.2 Security
9.3 Unique Safeguards and/or Security Features

In general, PeLUit-40 design development will accommodate existing safeguard and security guidance from IAEA via the review and design approval processes by BAPETEN.

As part of the safeguard aspect in PeLUIt-40 general scheme indicating the Key Measurement Points (KMP) and the fuel flow already developed as shown in below figure. It cover the flow of fresh fuel from outside to the system of PeLUit-40 and its related measurement of its quantity.Flow of spent fuel outside the PeLUit-40 and measurement of remaining spent fuel in the storage. The other internal flow are from reactor core to the damage fuel line including its measurement, and from the core to the spent fuel storage (also accommodating the option of re-circulated to the core for multi-pass option. I&C related with the measurements of each KMP are including in the general I&C system of PELUIt-40.

10. Project Delivery & Economics

SUMMARY FOR BOOKLET

10.1 Project Preparation and Negotiation
10.2 Construction and Commissioning
10.3 Operation and Maintenance