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Design Study for KALIMER Upper Internal Structure and Reactor Refueling System

KALIMER 노상부구조물 및 핵연료교환 시스템의 설계 연구

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제 출 문

한국원자력연구소장 귀하

본 보고서를 “Design Study for KALIMER Upper Internal Structure and Reactor Refueling System”에 대한 기술보고서로 제출합니다.

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국문 요약

KALIMER 노상부 구조물(Upper Internal Structure) 및 핵연료 교체 시스템에 대한 설계 연구를 수행하였다. 두 시스템의 뚜렷한 특징으로서는 착탈식 노 상부 구조물의 개념 도입 및 이로 인한 핵연료 교체 기간의 증가이다. 노 상부 구조물에 대해서는 기능, 구조, 재료 측면에서의 설계 요구 사항 정립 및 기능적 요건의 수용을 위한 개념적인 접근 방법을 도출하였다. 핵연료 교체 시스템의 경우에는 기능, 구조, 공정 및 I&C(Instrument and Controls) 측면에서의 설계 요구 사항을 마련하였으며 기능 및 공정 요구 사항을 수용하기 위한 방안을 개념적으로 정리하였다. 아울러 핵연료 교체 기간 증가로 인한 발전소 활용도(availability)에 대한 영향을 GE사의 Markov code를 이용하여 평가하였다.

착탈식 노 상부 구조물의 경우에는 착탈을 위한 부가적인 기구 및 캐스크 설계 문제가 해결되면 설계 요구 사항을 대부분 만족시킬 수 있을 것으로 판단되며, 핵연료 교체 시스템의 경우에는 기능 및 공정상의 설계 요구 사항은 IVTM 캐스크 및 이동 관련 계통의 추가를 고려하면 설계상으로 특별한 문제점 없이 만족될 수 있을 것으로 사료된다. 그리고 노 상부 구조물의 착탈로 인한 핵연료 교체 기간의 증가가 1 주일을 초과하지 않을 경우, 발전소의 활용도에 대한 영향은 무시할 수 있는 것(1% 이내)으로 나타났다.

Abstract

The design study for the KALIMER upper internal structure(UIS) and reactor refueling system has been described. Two distinct features are plug-in UIS and extended refueling outage. For the UIS system, the functional, structural and material requirements have been determined and the accommodation approaches to meet these functional requirements described. For the refueling system, the functional, structural, process and I&C (Instrument and Control) requirements have been established and the accommodation approaches for the functional and process requirements described. The impact on plant availability due to extension of the refueling outage has also been investigated.

The accommodation approaches for UIS system show that the design concept of the system will satisfy the functional requirements with a few design issues to be resolved, such as UIS plug in/out handling system and cask design. It is also shown that the functional and process requirements of the refueling system are achievable with the design of the IVTM cask and related transfer system and the extended refueling outage has little effect(within 1 %) on the plant availability if extra refueling time do not exceed 1 week.

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1 Introduction

The design study for the KALIMER upper internal structure(UIS) and reactor refueling system is described. Two distinct features underlined in this study are the plug-in UIS and extended refueling outage. The plug in/out UIS with a simplified fixed plug, to which UIS main cylinder is attached, is removed and replaced at each refueling outage with an in-vessel transfer machine(IVTM). The extended refueling intervals provide time for the UIS plug in/out process after reactor shutdown. For the UIS system, the functional, structural and material requirements are summarized and the accommodation approaches to meet these functional requirements are described. The accommodation approaches to meet the structural and material requirements will be done during the conceptual design stage based on the structural and thermal hydraulic evaluation results. The economic assessment will also be implemented during the conceptual design stage.

For the refueling system, there are the functional, structural, process and I&C (Instrument and Control) requirements. The accommodation approaches for the functional and process requirements are described in this report. The approaches for the structural and I&C requirements are briefly addressed in this report and will be covered more fully during the conceptual design stage. The impact on plant availability due to extension of the refueling outage is also investigated.

2 Reactor Upper Structure Simplification

2.1 System Description

The upper internal structure (UIS) is welded to the under side of the fixed plug and extends downward into the hot pool to near the top of the core. The principal functions satisfied by the UIS are: (1) lateral support of the control rod drivelines, (2) protection of the drivelines from sodium flow induced vibration, and (3) support of the above core instrumentation drywells. The UIS with the fixed plug is removed for reactor refueling to make room for the in-vessel transfer machine (IVTM). A transfer cask and adapter arrangement are used to remove the UIS and install the IVTM.

The UIS is shown in Figure 1~ 2. Principal features of the UIS are the fixed plug, the control rod driveline schroud tubes, the instrumentation drywells including the conduit and ducting used for their support, several horizontal structural plates, the structural cylinder, and insulation and shielding in the region below the reactor closure. The UIS main cylinder overall length is 865 cm(TBU) and its diameter is 240 cm(TBU). The rod shroud tubes extend from the fixed plug to within 5.0 cm (TBU) of the top of the core assemblies during reactor power operation.

UIS Main Cylinder

The principal structural member of the UIS is the 240 cm (TBU) in diameter SS-316 cylinder that extends down to approximately 90 cm (TBU) above the core outlet. The cylinder's wall thickness of 2.5 cm (TBU) was selected to assure adequate resistance to seismically induced displacements thus satisfying requirements on the motions of the control rod drivelines relative to the core. Additional stiffness for the cylinder is obtained from the three horizontal plates that are welded to its walls and except for penetrations for the shroud tubes, span the area inside of the cylinder. The bottom end of the structure is provided with two liners that protect it from the thermal environment at the core outlet. The outermost liner is made from Inconel Alloy 718 and is used for thermal striping thermal protection. The second SS-316 liner, in conjunction with outer liner is used to insulate the structure against rapid temperature changes occurring during scram transients.

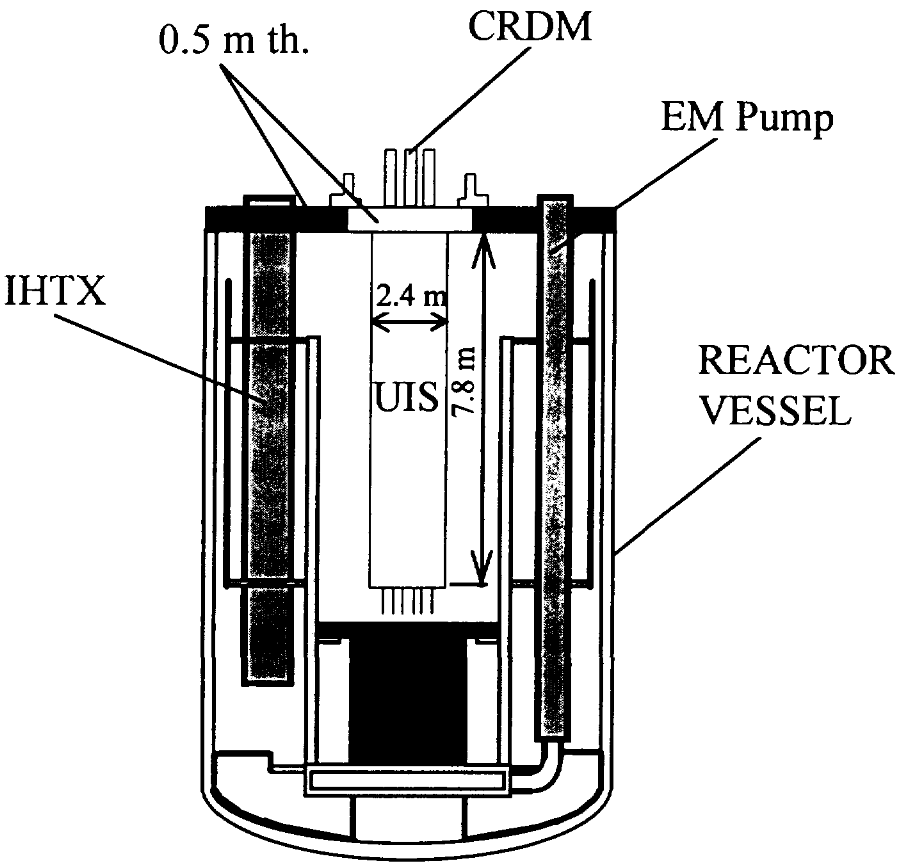


Figure 1 Reactor Internals Overview

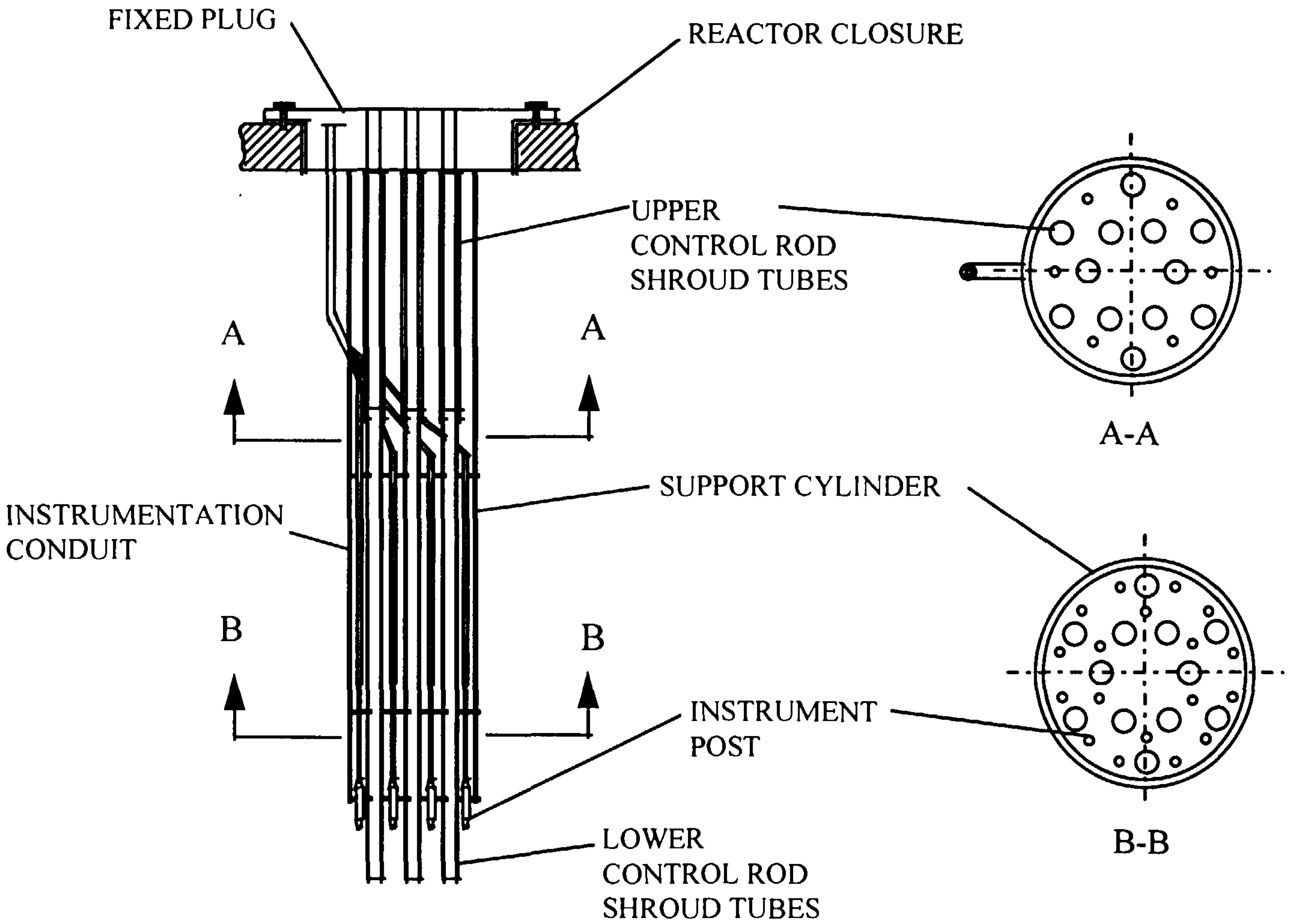


Figure 2 Upper Internal Structure

Shroud Tubes

There are nine shroud tube assemblies, for each of the nine control rod drivelines. Each assembly consists of an upper SS-316 tube(TBU), a lower Inconel Alloy 718 tube(TBU), and an internal bushing as shown in Figure 3. The tubes are sized and located such that the drivelines pass through the center of the assemblies without contacting the tubes except over a region near mid-elevation where the close fitting guide bushing is located. As shown in Figure 4, the Inconel Alloy 718 bushing is positioned within the tube by a series of close fitting diameters and a shoulder that supports it in the vertical direction. It is mechanically attached to the upper SS-316 shroud tube with a SS-316 ring which is pinned and welded to the tube body. The upper SS-316 tube itself is welded to the horizontal structural plate near elevation 435 cm(TBU). From here it extends upward to the fixed plug and downward to the lower horizontal structural plates. The upper end is captured in a nozzle protruding from the underside of the plug. The lower end is similarly captured in the lower Inconel Alloy 718 tube which, as shown on Figure 5, is mechanically attached to the lower horizontal structural plates of the UIS. This arrangements minimizes the thermal interaction between the shroud tubes and the UIS structurals due to differential temperatures and materials and eliminates the need to use dissimilar metal welds. Inconel Alloy 718 was selected for the lower tube to sustain the thermal striping and thermal shock conditions existing near the core outlet.

Instrumentation Drywells

There are twenty (TBU) drywells routed from the top of the UIS plug to the region directly above the reactor core. These pass through and are supported by the UIS. Each SS-316 drywell is 1.25 cm in diameter and will carry multiple thermocouples. The lower end of the drywells, which are located approximately 45 cm above the core outlet, are contained in heavy Inconel Alloy 718 forgings which provide structural support and thermal protection. The most severe mechanical loads would be due to the sodium flow induced vibration. The Inconel Alloy 718 material is specified to guard against thermal fatigue that would occur due to the steady state thermal striping and the thermal shock arising during scram transients.

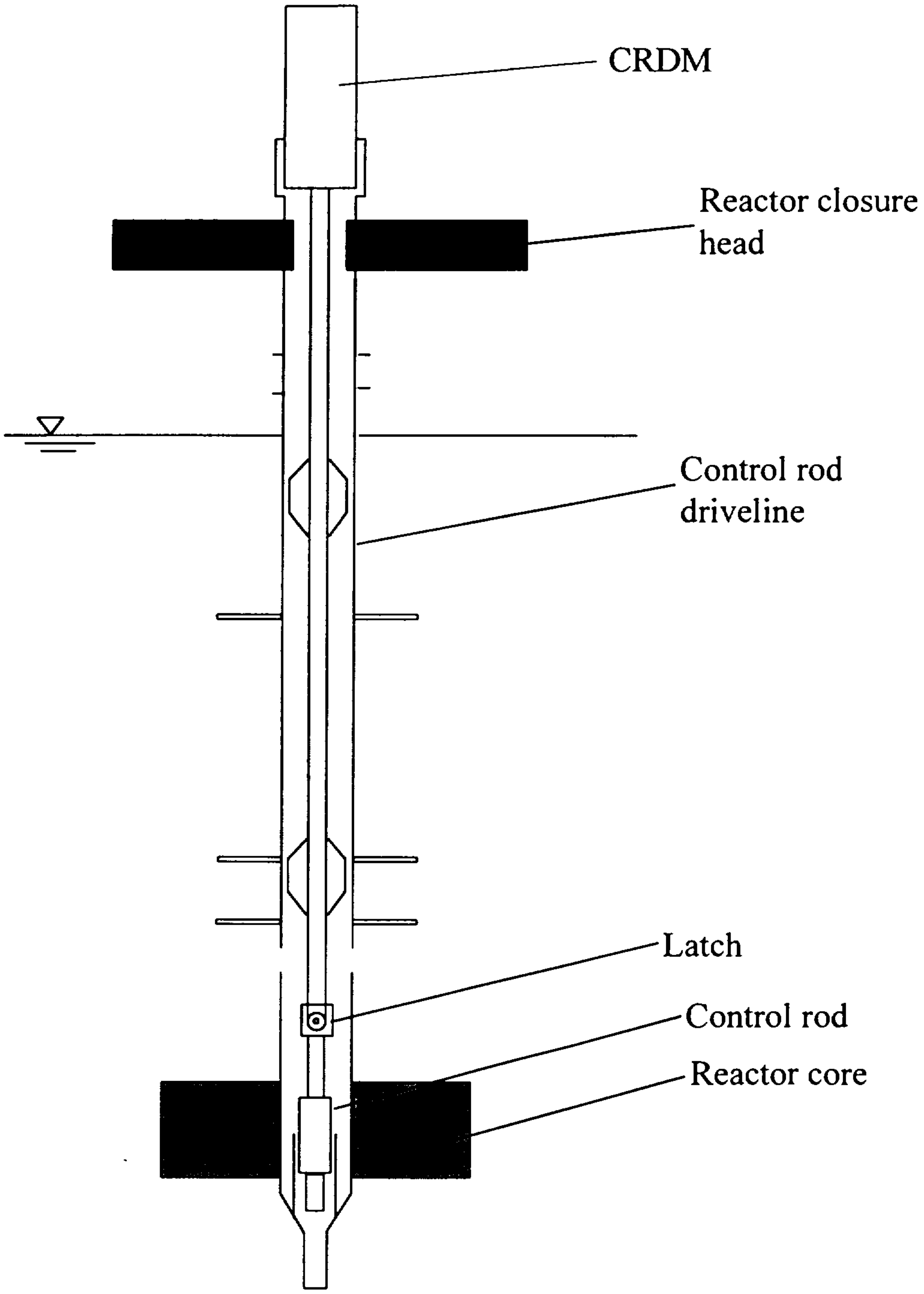


Figure 3 Section view of shroud tube and control rod

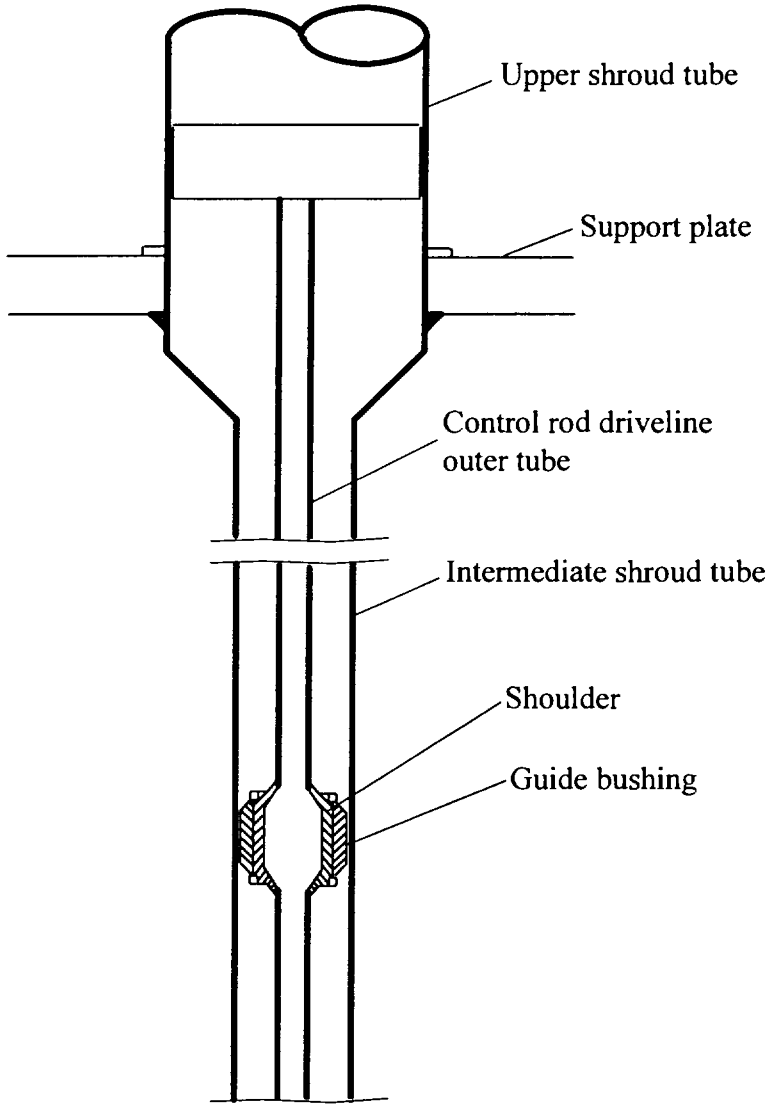


Figure 4 Upper Shroud Tube Details

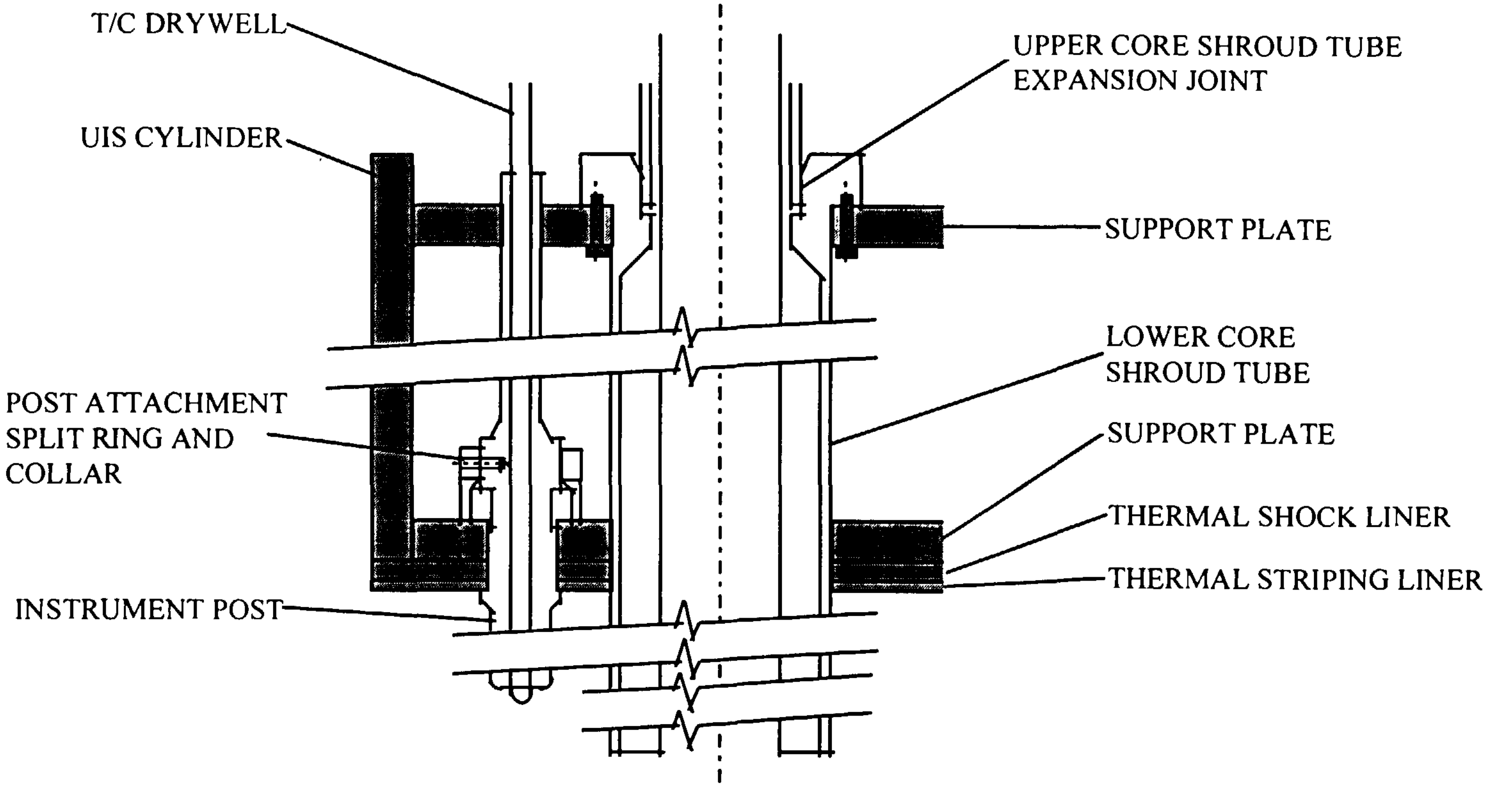


Figure 5 Upper Internal Structure Detail

The drywells are evenly distributed over the core outlet plane so as to provide information on the various core regions. Above this plane they are routed so as to be allowed to exit the reactor through a single port, thus the number of required plug penetrations is minimized. Between the port and the instrument post the drywells are enclosed in conduits and ducts that facilitate their routing and provide mechanical protection.

Fixed Plug

The fixed plug is part of the reactor closure. It supports the UIS, the control rod drive mechanisms and the ultimate shutdown mechanisms. In the center area of the fixed plug are nine openings for the control rod drive mechanisms and the ultimate shutdown mechanisms. There is also one penetration for the above-core instrumentation and a penetration for an access port for reactor internal inspection and maintenance. Attachment and support of the fixed plug to the closure is with a stepped shoulder arrangement and bolts. The stepped rims minimize radiation streaming. Sealing of the fixed plug is with multiple o-ring with buffered gas. Handling attachments are located at the top of the fixed plug for removal of the plug/UIS during refueling.

2.2 Design Basis

2.2.1 Functional Design Requirements

a) The UIS and fixed plug shall provide positioning and support for the reactor control and shutdown system components and the above-core instrumentation.

b) The UIS and fixed plug shall be designed to be removable during refueling and maintenance activities when the UIS and fixed plug are replaced with a plug-in type IVTM assembly(IVTM and another fixed plug).

c) The UIS shall provide protection for the control rod drivelines to prevent damaging due to vibrations by crossflow of the core coolant or seismic events.

d) The UIS and fixed plug in conjunction with the reactor closure and reactor support shall limit the horizontal deflection and acceleration to within the capability of the control rod driveline.

e) The fixed plug in combination with the reactor closure and reactor vessel shall provide the primary boundary for the primary coolant and its cover gas.

f) The UIS and fixed plug in combination with the reactor closure shall provide sufficient radiation shielding for personnel access to the head access area during power and shutdown operations. The shielding shall meet the ALARA personnel exposure requirements of 10CFR20 and 10CFR50 Appendix 1 and shall include the effects of generation, distribution and plate-out of radioactive fission products and corrosion products.

g) The fixed plug in combination with the reactor closure shall be designed to limit the influx of air into the cover gas to a rate less than 5 cc/day ($3.5 x 10^{-3}$cc/min) at standard temperature and pressure under all design conditions of reactor operation and shutdown, except during refueling and maintenance.

2.2.2 Structural Requirements

a) Duty Cycle

The UIS and fixed plug shall be designed to withstand the core outlet flows and local temperature variations which will be specified by other design group(TBD*). The reactor system shall be designed to sustain the temperatures, pressures and forces associated with the

*TBD:To be determined later

duty cycle events defined in Tables 1, supplemented with the fabrication, handling, transportation and installation loads. In addition to the classification of the different operating conditions in Service Levels A, B, C, and D as described in Table 1, components designed to satisfy the ASME Code may require definition of Design Condition in which case the Design Condition shall envelope the most severe steady-state operating pressures and temperatures.

Table 1 Reactor Module Operational Requirements

Reactor Module Power*	840 MWt
Reactor Module Design Life	60 years
Primary Coolant	Sodium
Primary Cover Gas	Helium
Core Bulk Outlet Temperature*	TBU (499°C)
Core Bulk Inlet Temperature*	TBU (360°C)
Core Delta T*	TBU (139°C)
Thermal Stripping Potential	TBU (150 °C)
Primary Circuit Delta P	TBU (100psi) nominal, TBD (115 psi) max
Primary Sodium Bulk Temperature (Long Term)	
Level A Events	TBU (499°C) max
Level B Events	TBU (593°C) max(<100 hrs)
Level C Events	TBU (677°C) max(<2000 hrs)
Level D Events, anticipated transients without scram to be accommodated 1) Loss of primary flow without Scram 2) Loss of Heat Sink Without Scram 3) Transient Over Power Without Scram (withdrawal of all control rods under the control of the plant control system)	Peak Mixed Mean Coolant Outlet Temperature <1 Hour TBU (760°C) >1 Hour TBU (733°C)
RVACS Operation (Level C), Peak Hot Pool Temperature	TBU (677°C) max
Reactor Refueling Temperature	400 °C
Reactor Hot Standby Temperature	550 °C
Reactor Cover Gas Pressure*	1.0 atm

*Nominal value at 100% power.

The loadings shall include, but are not limited to, gravity, internal and external gas and sodium pressures, hydrodynamic loads including flow induced vibrations, differential thermal expansions resulting from differences in material behaviors and steady-state and transient temperature gradients, irradiation-induced volume changes and deformations, support reactions and component interactions, sodium-water reactions, and seismic loads. The loads shall be established through analysis of system and component responses to the duty cycle events and the enveloping operating conditions described in Table 1. Different loading conditions in the duty cycle may be grouped and enveloped by umbrella events provided the enveloping events are at least as severe as the umbrellaed events in magnitudes, durations, and number of cycles, and the grouping accounts for any sequencing effects. The calculated component loads shall be included in the component equipment specification, and the design of each component shall be demonstrated through analysis or test to satisfy the applicable structural and functional design criteria.

b) Seismic Criteria

All structures and components of the UIS and fixed plug in combination with the reactor system shall be capable of withstanding the effects of the Operating Basis Earthquake (OBE) without loss of capability to remain functional, and to withstand the effects of the Safe Shutdown Earthquake (SSE) without loss of capability to perform their safety functions.

OBE/Plant Condition Load Combinations

The reference OBE horizontal and vertical maximum ground accelerations are 0.15 g. The free field response spectra are as defined by NRC Regulatory Guide1.60. In addition the reactor shall be designed to withstand a TBD g OBE.

Five OBEs, each with ten maximum peak response cycles, shall be assumed to occur over the design life of the plant. Four of these OBEs shall be assumed to occur during the most adverse normal operating conditions (Level A Service Limit) determined on a component limiting basis. The other OBE shall be assumed to occur during the most adverse upset event (Level B Service Limit) determined on a component limiting basis, and at the most adverse time in the upset event.

SSE/Plant Condition Load Combinations

The reference SSE horizontal and vertical maximum ground accelerations of 0.3 g are based upon the ALMR siting envelope criteria. The free field design response spectra are as defined by NRC Regulatory Guide 1.60. In addition, the reactor shall be designed to withstand TBD g peak ground accelerations.

One SSE, with 10 maximum peak response cycles, shall be assumed to occur over the design life of the plant. This SSE shall be assumed to occur during the most adverse normal operation (Level A Service Limit) or upset (Level B Service Limit) event determined on a component limiting basis. During and following the SSE the pony motors on the intermediate pumps are assumed to be functioning.

Design Criteria

Design of the UIS and fixed plug in combination with the reactor system components shall conform to the design codes and standards. Establishment of the permissible values of stresses in applying these codes and standards shall allow for any known or predictable degradation of material performance that may occur over the design life as a result of exposure to sodium, irradiation, and stress at service temperatures. The reactor system components, structures and subsystems shall be designed as Seismic Category I structures.

In addition to the stress and strain limits established by the design codes and standards listed in Table 2, the design shall satisfy appropriate deflection limits to assure required functional performance. These shall include limits on control rod lateral deflection during seismic loading such that reactivity insertion from core/control rod seismic separation is limited.

c) 60-Year Design Life

- The UIS and fixed plug shall be designed for a service life of 60 years.

Table 2 Component Codes and Standards for Design and Construction

Component	Code
Reactor Fixed Plug	ASME Section III, Subsection NB with High Temperature Code Cases; 10CFR100
Reactor Internal Structures	ASME Section III, Subsection NG with High Temperature Code Cases

2.2.3 Material Requirements

a) The materials of construction of the UIS and fixed plug components shall be selected on the basis of performance in fast reactor and liquid sodium environments. Constituent elements whose transmutations have long half-life's shall be controlled to minimize their impact on disposal costs and methods.

b) The effects of environmental conditions such as neutron radiation exposure, temperature, and sodium shall be included in determining the allowable value of material properties used in the design of system components.

c) Material surfaces in contact with liquid sodium coolant shall be austenitic stainless steel unless other materials must be used for strength, thermal striping resistance, wear resistance, or reduced irradiation swelling.

d) Surfaces that experience relative motion during operation, installation, or removal shall be made of suitable material combinations or shall be provided with hard-surfaced regions to provide adequate wear properties and to preclude galling or seizing.

e) Appropriate heat treatments and processes shall be utilized during component fabrication to minimize sensitization.

2.3 Accommodation Approaches to Meet Functional Requirements

In this section, the functional estimation of the UIS and fixed plug are described to show that the design described in Section 2.1 meets the requirements set forth in Section 2. The design requirements can be separated into two area, functional and structural. The ability of the UIS and fixed plug to satisfy structural requirements consisting mainly of stress, strain and deflection limits is covered in Sections 4.3 and 4.5. Here the structural performance of the UIS and fixed plug is described plus its ability to satisfy stress, strain and deflection limits. The structural evaluation of the UIS consists of detailed finite element modeling to calculate stress, strain and deflection levels which are then compared with the stress and strain limits specified in the structural design criteria and the deformation limits derived from the functional requirements.

The ability of the UIS and fixed plug design to meet functional requirements is described in the paragraphs following. This evaluation consists of discussing the UIS and fixed plug features that satisfy a particular function. Since the KALIMER is presently at the preliminary design stage, the functional evaluation is at a general level since specific details are not yet developed. In some cases the KALIMER design is compared to PRISM to obtain an indication of meeting functional requirements since the two plant design are similar.

A) Positioning & support of reactor control and shutdown systems and above-core instrumentation:

This requirement is satisfied by incorporating shroud tubes and drywells into the UIS design as shown in Figure 2. Support and positions of the reactor control and shutdown systems are provided by the 12 shroud tubes and the 12 stub tubes. The stub tubes provide vertical and lateral support for the control rod drive mechanism above the fixed plug and the shroud tubes lateral support for the drivelines between the fixed plug to the top of the control assembly. The shroud tube locations correspond to the locations of the control assemblies in the core. The shroud extents from the underside of the fixed plug to a few centimeters above the control assembly. This provides a continuous cylinder to laterally position the driveline during reactor operation yet accommodate differential expansions between the driveline and the UIS. The shroud tube design allows the driveline to be withdrawn for maintenance or replacement. The diameter and position of the shroud tube are such that the driveline is funneled into the opening of the control assembly during driveline installation.

Four drywells are included in the UIS for monitoring of the core mixed outlet sodium temperature. The drywells extend a few centimeters from the underside of the UIS and are equally spaced near the periphery of the UIS main cylinder. Each drywell extends from the underside of the UIS to above the fixed plug. The tube from each of the four drywells merge into a single tube underneath the fixed plug resulting in one instrument penetration for the fixed plug. Each drywell inner diameter is sized for 5 thermocouples, one for each of the four RPS channels plus one spare. The drywell is sealed and pressurized with an inert gas to provide an inert atmosphere for the instrument sensor during the high temperature operation. This also acts as a buffer seal in the event the drywell develops a leak.

B) Removable during refueling & maintenance

For refueling, the UIS and fixed plug assembly is removed from the reactor and replaced by a plug-in IVTM. To facilitate its removal and re-installation and to minimize the reactor outage time, the UIS and fixed plug assembly has certain features to facilitate its removal and re-installation. The entire UIS and fixed plug assembly is removed including the control rod assemblies and above-core instrumentation. Prior to the UIS and fixed plug removal, the control assemblies are unlatched from the driveline and the drivelines are withdrawn out of the control assembly into the shroud tube which clears the UIS of the core assemblies.

The fixed plug is bolted to the closure head during power operation. These bolts are readily accessible from the top of the closure to minimize the unbolting time. A stepped arrangement at the interface of the fixed plug with the closure head allows the fixed plug/UIS to be lifted clear of the closure head with straight vertical lift motion. Lifting eyes are also installed on the top of the fixed plug circumference for ready attachment to the lifting crane which draws the UIS/fixed plug into the UIS cask. These same features permit the UIS and fixed plug to be re-install into the UIS/fixed plug penetration after completion of refueling.

For re-installation, leading edges of the fixed plug are tapered inward to facilitate insertion and self centering of the fixed plug in the reactor closure opening. There is also a key arrangement for self-azimuth alignment of the UIS with the closure to ensure alignment of the shroud tubes with the core control assembly positions(See Figure 6).

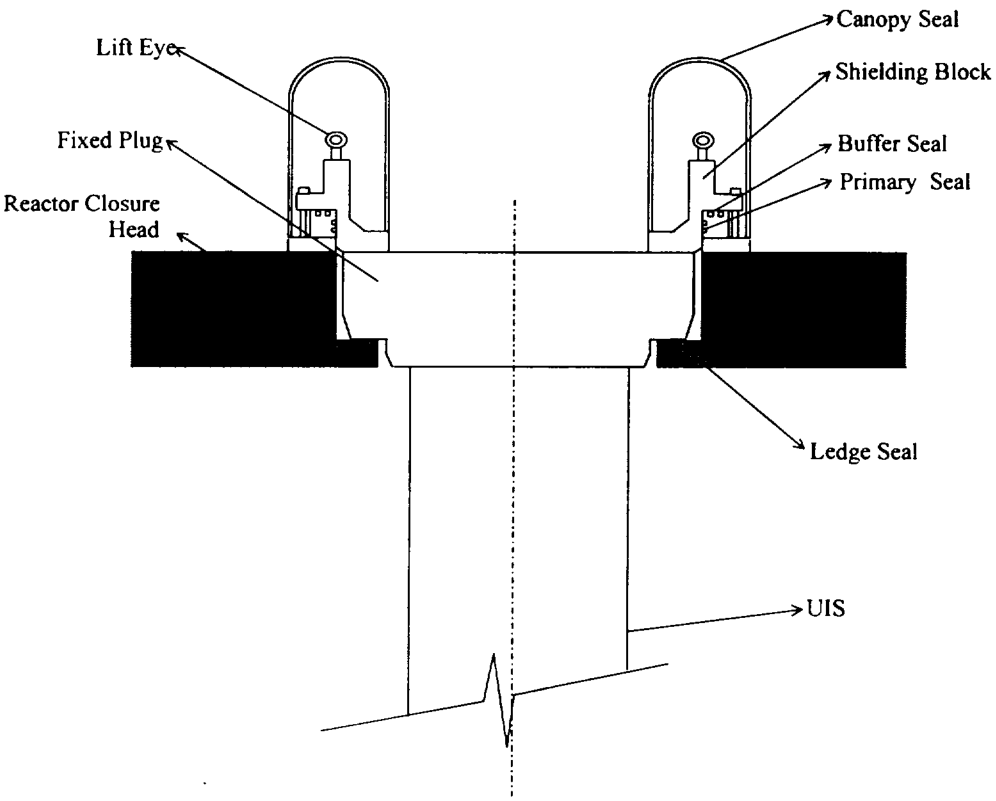


Figure 6 Removable UIS Fixed Plug

C) Protection for the control rod drivelines due to damaging vibrations

Possible sources of excitation of the UIS and associated structures are as follows:

- Excitation of the UIS as a whole from impingement of the core exit sodium flow on the UIS bottom plate.
- Lower shroud tube excitation from vortex shedding in the coolant cross-flow in the core exit region.
- Instrument drywell excitation from vortex shedding in the coolant cross-flow in the core exit region.

The UIS and control rod drivelines have the potential for FIV (flow-induced vibrations) due to the core exit flow on the UIS bottom plate. The fluid-structure interaction and/or resonance between a structural natural frequency and characteristic frequency associated with fluid flow could lead to large amplitude motion and severe vibrations. This vibrations could lead to fatigue type failures of the UIS structures and severe rattling of the driveline and control rod. To prevent the UIS from damaging due to FIV, the thickness in conjunction with length and diameter will be determined so that the natural frequencies of the shroud tube and driveline are well separated from the driving frequencies.

Two features in the shroud design prevent damaging vibrations of the driveline and control bundle. The shroud tube encloses the driveline which isolates the driveline from the cross flow above the core. In addition, guide tube bushings are axially installed inside each shroud tube resulting in a close sliding fit with the driveline which will prevent large amplitude motion of the control drivelines due to the damaging vibrations. The guide tube bushing arrangement is shown in Figure 4.

To prevent FIV of the UIS as a whole, the lower guide tube and the lower instrument drywells, vortex shedding frequency analysis and flow impingement analysis will be required to show that their natural frequencies are well separated from the driving frequencies.

Cross-flow for the driveline is not expected to be a concern since the driveline is enclosed by the shroud tube and unshrouded for only 50 mm immediately above the core assemblies.

Typical natural and flow excitation frequencies for the KALIMER UIS are estimated assuming every structure as a cantilever and in-sodium natural frequency is approximately 1/3 of in-air case which makes the natural frequency come close to the excitation frequency. The result shows adequate separation and very low potential for FIV as shown below.

Structure/Feature	Calculated natural frequency	Calculated excitation frequency
Entire UIS	11 Hz	~0.1 Hz
Lower shroud tube	54 Hz	1.1~1.7 Hz
Lower instrument drywell	98 Hz	3~4.5 Hz

D) Core/control rod drivelines/UIS horizontal relative deflection limit

For the KALIMER reactor, the core is supported from the bottom of the reactor vessel and the UIS from the reactor closure head. Relative lateral deflection of the UIS and core must be limited to maintain the scram functions during OBE and SSE vents. The requirement is to operate through an OBE event without spurious release or scram of the control rods. Release of the control rod from seismic loads during an SSE event is permitted. However, the UIS/core relative deflections must not damage the scram function of rod insertion by driveline drive-in of the absorber bundle. Then the UIS/core relative deflections could damage the control rod drivelines and interfacing components. This could affect the subsequent control rod manipulations and may prevent driveline insertion to follow the control rod into the core. So the capability to preclude such interference is required to assure scram in the case of a control rod being stuck in the latch or the control assembly. The relative UIS/core deflection limit is expressed as:

$$S = C + D - M,$$

where S = seismic deflection limit

C = inter-element clearances

D = permissible driveline and shroud tube deflections

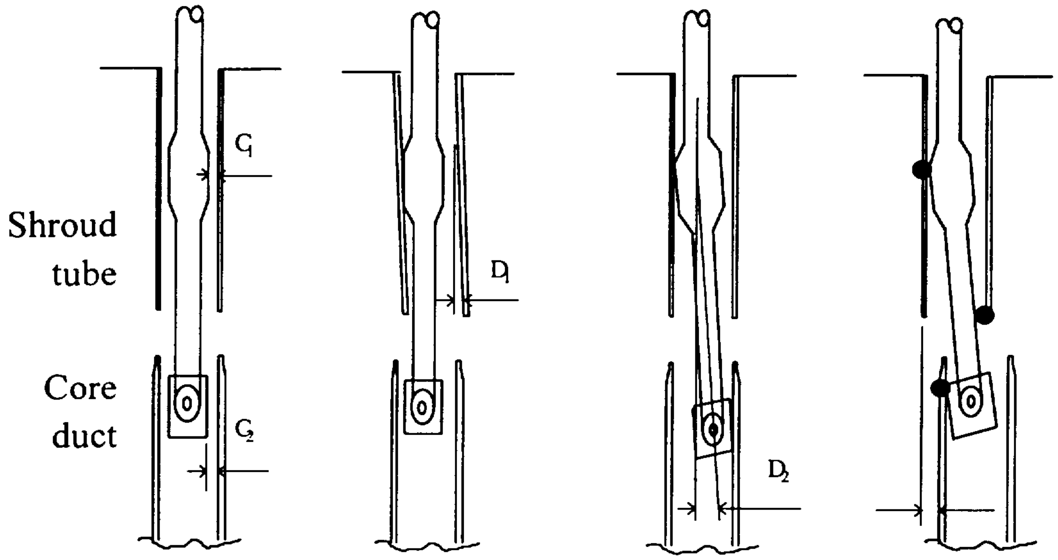
M = misalignment allowance.

The misalignment will be primarily determined by the system tolerance stack-up, core porosity and deformations, and operating thermal gradients with little contribution from the driveline or shroud tube. Therefore the misalignment M has to be accommodated rather than controlled in the driveline design. The maximum expected misalignment M and the maximum permissible clearance C would determine the displacement at which the UIS and the control assembly just contact the driveline on opposing sides (See Figure 7). Any UIS/core relative displacement beyond such 'two-point contact' would deform the driveline and possibly the shroud tube and should be limited to the deflections which would not permanently deform the components (OBE requirement) or deflections where the friction at interference points would not prevent driveline insertion into the core(OBE or SSE requirement).

In case of KALIMER, the inter-element clearances(C) and permissible drivelines and shroud tube deflections(D) will be evaluated based on the structural data which will be determined later. The misalignment allowance(M) should be estimated based on core/thermal hydraulics analyses which will also be performed later. Then the seismic deflection limit could be determined and compared with the maximum deflection resulted from a seismic analysis. If the maximum deflection exceed the limit, the parameters such as inter-element clearances, shroud tube dimensions, and misalignment allowance are modified and seismic analysis performed, repeatedly.

E) Primary boundary for the primary coolant and cover gas

The primary boundary is maintained by the fixed plug. Sealing of the fixed plug to the closure head is accomplished by the ledge and buffer seals as shown in Figure 7. The ledge seal is essentially two flat horizontal surfaces in contact to interrupt sodium vapor passage in the reactor cover gas from reaching the upper portion of the closure. However, this can not be a sufficient means to make the primary boundary. Thus the buffer seals are additionally installed on the interfacing surfaces between the reactor closure head and the fixed plug shielding block.



Where,

S=C+D-M

C: Inter element clearance

=Max[C1,C2]

D: Shroud tube & driveline deflection

=Max[D1,D2]

M: Misalignment allowance

Figure 7 Relative Horizontal Deflection

Figure 8 illustrates the principle of the buffer seal. The buffer seal separates the interfacing space into three regions. The first is outside part over which the atmosphere pressure exerts. The second region is a cover gas region in which the pressure level is slightly higher than that of the atmosphere. The third is the space between the outside and the cover gas regions. This region separates the first and second regions using two O-rings and supplied Helium gas which has a slightly higher pressure level than the cover gas region(the second). Since the Helium gas in the second region has the highest pressure among three regions, there is no possibility for the cover gas or sodium coolant to leak through the primary boundary.

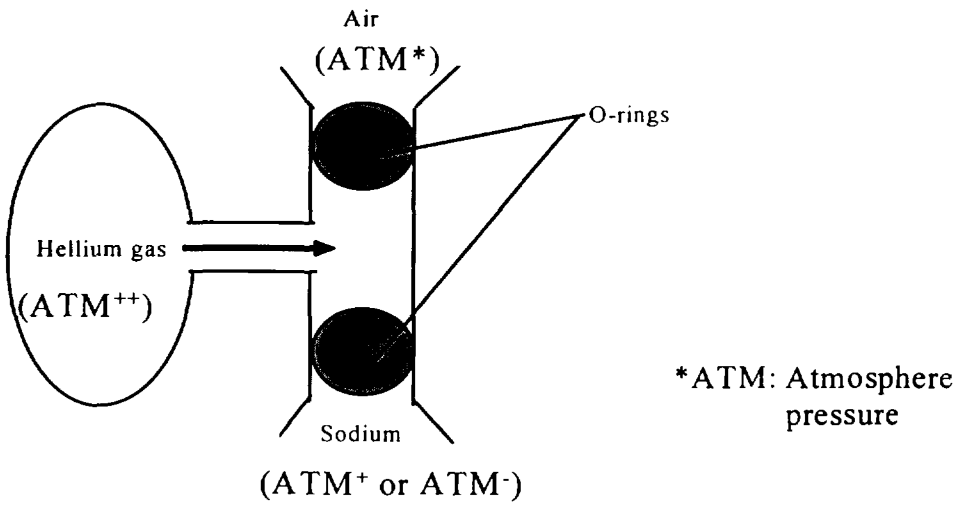


Figure 8 Buffer Seal

F) Radiation shielding for personnel access

The area above the reactor closure is designated as the head access area. Plant personnel have need to enter this area during all phases of plant operation. Shielding is required to meet ALARA goals and meet annualized personnel radiation exposure limits. The head access area is designated a plant radiation zone II which requires that dose rate to be limited to 2.5 mrem/hr(see Figure 9). Personnel access to the head access area is limited to 40 hr/week.

To meet this exposure limit, shielding is required at the reactor closure, the component penetrations, instrument penetration and the UIS/fixed plug penetration. For the UIS and fixed plug design, this requirement is satisfied by the thickness of the fixed plug, the fixed plug to closure head penetration, the control drive mechanism penetration and the instrument penetration.

Shielding by the fixed plug is maintained as follows:

- The fixed plug thickness is the same as the reactor closure head, Figure 10.
- The thermal insulation plates under the reactor closure are maintained under the fixed plug offering the shielding benefit of this additional steel.
- The fixed plug-closure head interface face is stepped and the gaps are covered with additional above-head shielding to minimize radiation streaming, Figure 10.
- The control rod drive penetrations and instrument penetration are fitted with a shield plug to provide the equivalent steel thickness of the closure and thermal insulation plates and are stepped, all to minimize radiation streaming, Figure 11.

G) Limitation of the air influx into the cover gas

Intrusion of air (a source of oxygen) into the inerted reactor system must be limited to minimize contamination of the sodium and limit the burden on the sodium and cover gas cleanup systems(see Figure 12). The requirement to limit air intrusion to less than 5 cc/day (3.5×10^{-3} cc/min) is based on staying within the capacity of the reactor and cover gas cleanup systems and prevent buildup of contaminant in the reactor.

Air in-leakage is mostly likely to occur through the reactor closure penetrations of which there are many. The reactor vessel which is welded structure is considered leak tight. Leakage tightness of the closure penetrations is maintained by the appropriate seal design. Estimated leakage through the closure penetration is summarized in the table below. It shows that estimated total leakage through the closure penetrations will be below the established limit of 5 cc/day (3.5×10^{-3} cc/min).

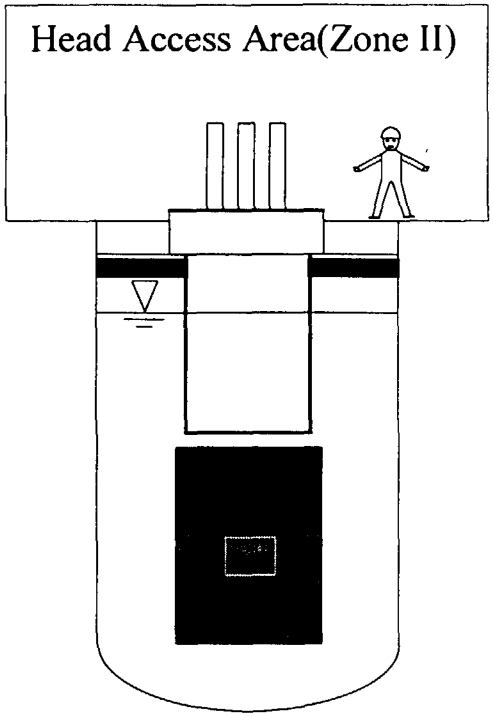


Figure 9 Head Access Area

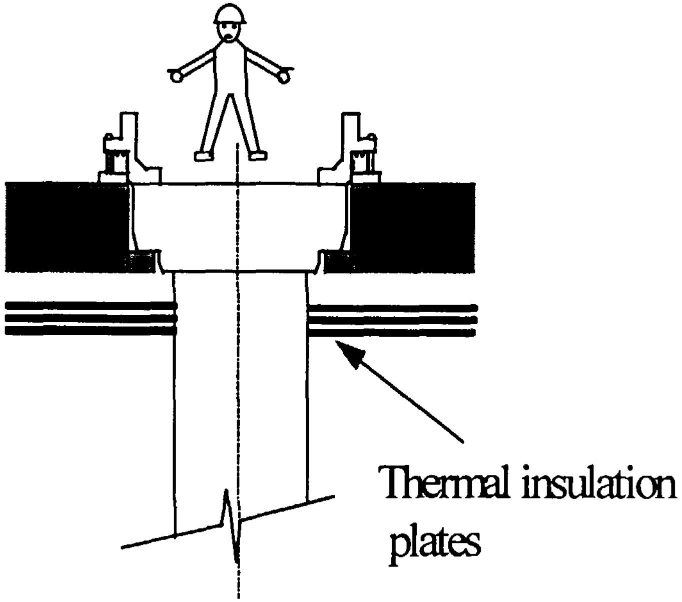


Figure 10 Radiation Shielding for Head Access Area

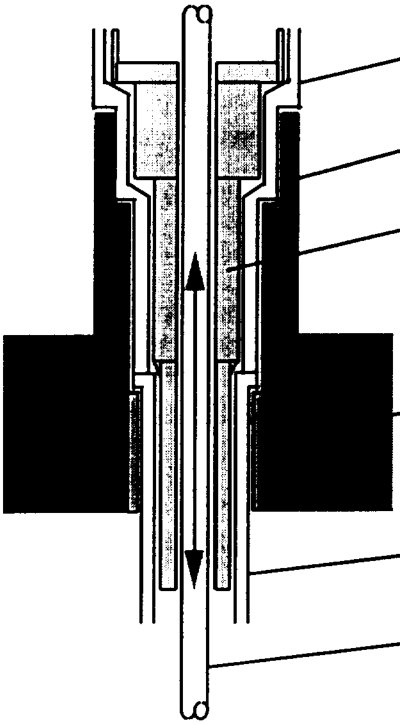


Figure 11 Shield Plug

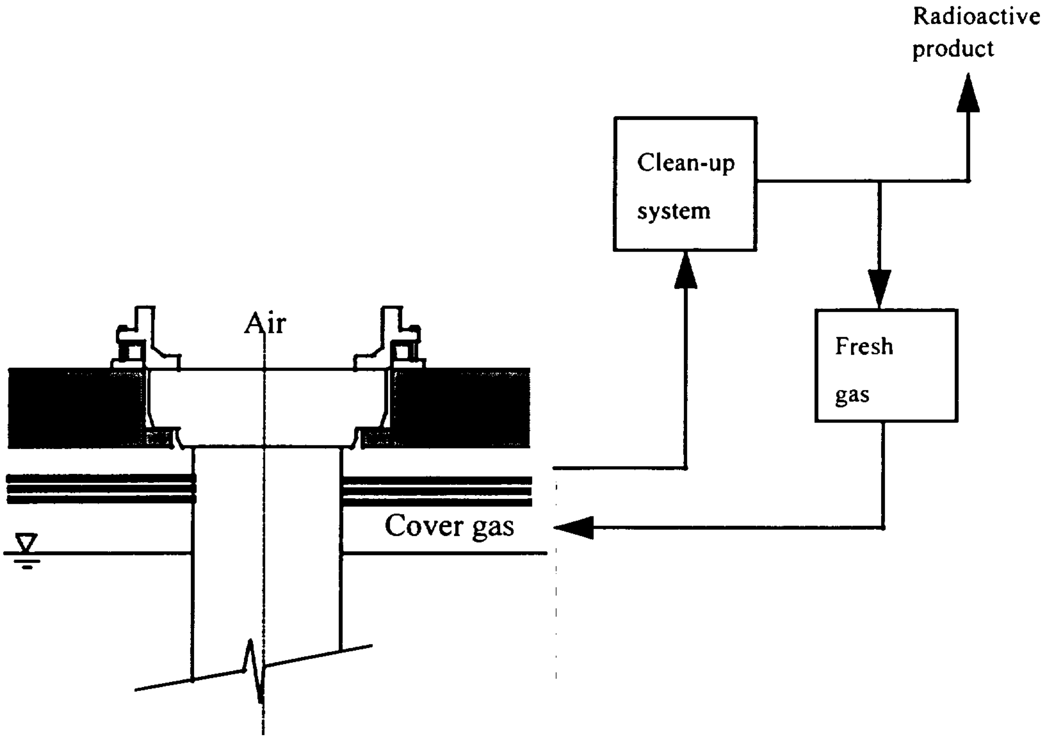


Figure 12 Conceptual diagram of Cover Gas Clean-up System

Closure Penetration	Type of seal	Estimated leak rate, cc/min	Number of penetrations	Total leak rate, cc/min
Fixed plug	Double elastomer seal	10^{-4}	1	$1.0\text{x}10^{-4}$
Control rod mechanism	Double metal seal	10^{-6}	12	$0.12\text{x}10^{-4}$
IHX	Metallic seal	10^{-5}	2	$0.2\text{x}10^{-4}$
EM pump	Metallic seal	10^{-5}	4	$0.4\text{x}10^{-4}$
Instrument	Double metallic seals	10^{-6}	10	$0.1\text{x}10^{-4}$
Others	Double metallic seals	10^{-5}	10	$1.0\text{x}10^{-4}$
			Total	$2.82\text{x}10^{-4}$

In addition, certain feature and operating modes will further limit air in-leakage for the reactor system. These are as follows:

- The reactor cover gas pressure will be only slightly below atmosphere to limit driving pressure differential.
- Most seals are double buffered seals which directs gas through a leaking seal toward the outside atmosphere and preventing air passage through leaking seal into the reactor.

3 Refueling System

3.1 System Description

3.1.1 Summary Description

The reactor refueling system (RRS) provides the means of transporting, storing and handling reactor core assemblies, including fuel, blanket, control, and shield, within the KALIMER plant. The system consists of the facilities and equipments needed to accomplish the normal scheduled refueling operations and all other functions incident to the handling of core assemblies. It also includes the equipments for removal of the UIS system and installation of the IVTM assembly for refueling.

The RRS comprises three subsystems:

1. Reactor Fuel Handling System (RFHS)
2. Transport System (TS)
3. Fuel Receiving, Storage, and Shipping System (FRSSS).

Figure 13~ 14 depicts the arrangement of the RRS and a conceptual fuel flow is shown in Figure 15.

The function of the RFHS is to refuel the reactor. The system consists primarily of the in-vessel transfer machine (IVTM) assembly, which is comprised of the IVTM and the non-rotating plug, and the fuel transfer port which are located entirely within the reactor area of the module.

The TS provides the means of moving the core assemblies between the reactor (RFHS) and the fuel cycle facility (FRSSS) during the refueling outage.

The FRSSS provides the means of receiving, storing, and transferring the core assemblies to the co-located fuel cycle facility. It also supports the RFHS during the refueling of the reactor.

New control and shield assemblies enter the fuel storage and transfer building (FSTB), are unloaded from their shipping containers, inspected, and are temporarily stored. New fuel and blanket assemblies are received from the co-located fuel cycle facility and stored in the fuel handling cell (FHC). After reactor shutdown for refueling, the UIS system is removed from the reactor closure head and transferred into UIS system transfer cask(UTC), and then the UTC containing the UIS system is moved into casks vault(CV) within the reactor head access area for temporal storage. Next, the IVTM assembly transfer cask(ITC) is transferred from the CV to the top of the reactor closure head and the IVTM assembly is installed in the same reactor head opening as for the UIS system. The conceptual flow diagram from UIS system plug-out to IVTM assembly plug-in is shown in Figure 16.

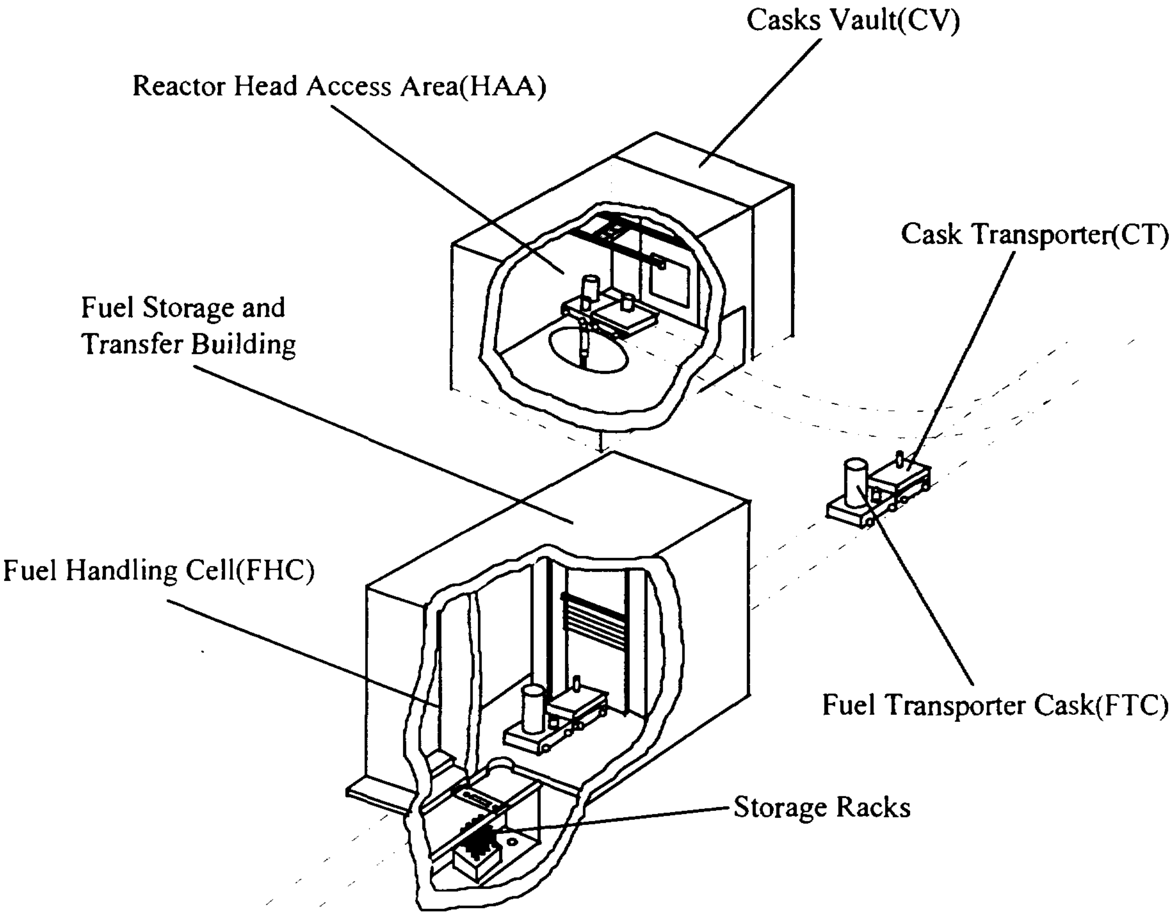


Figure 13 Reactor Refueling System(RRS)

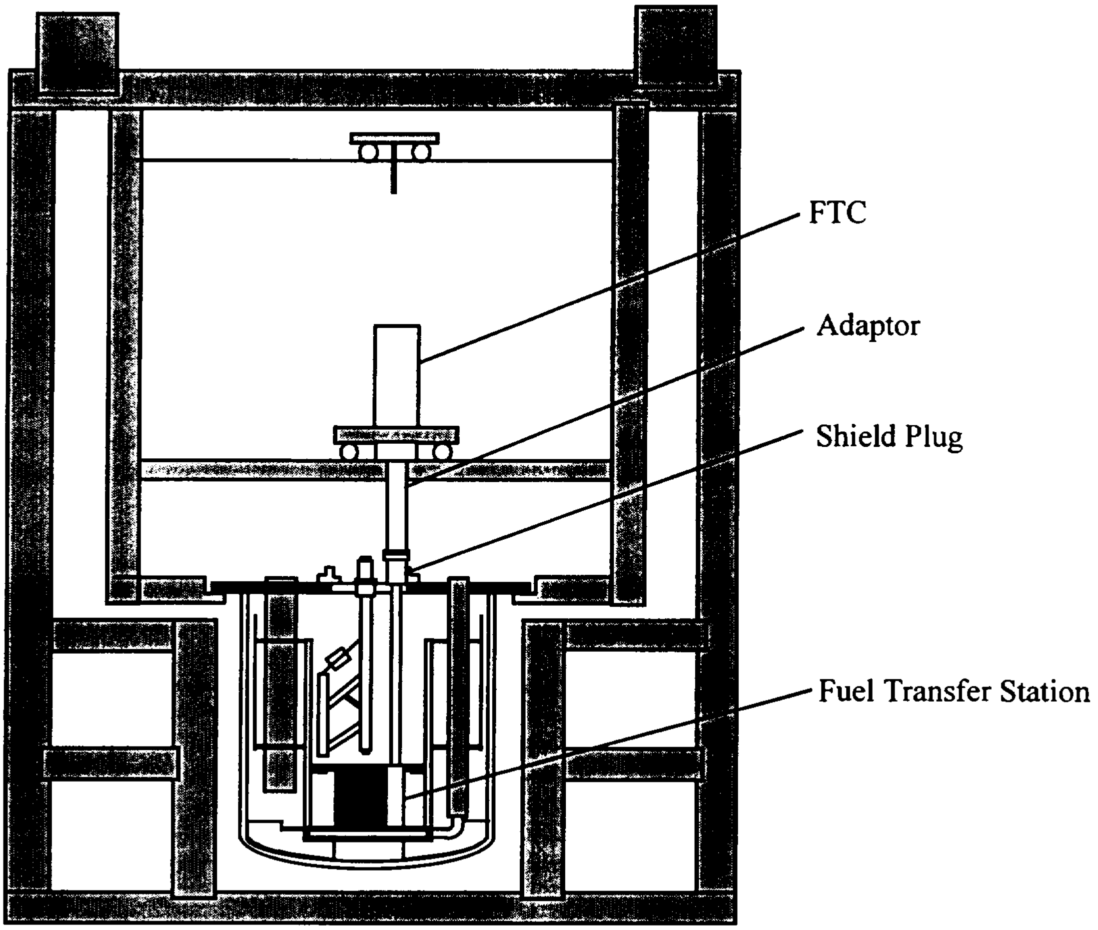


Figure 14 Section View of Reactor Refueling

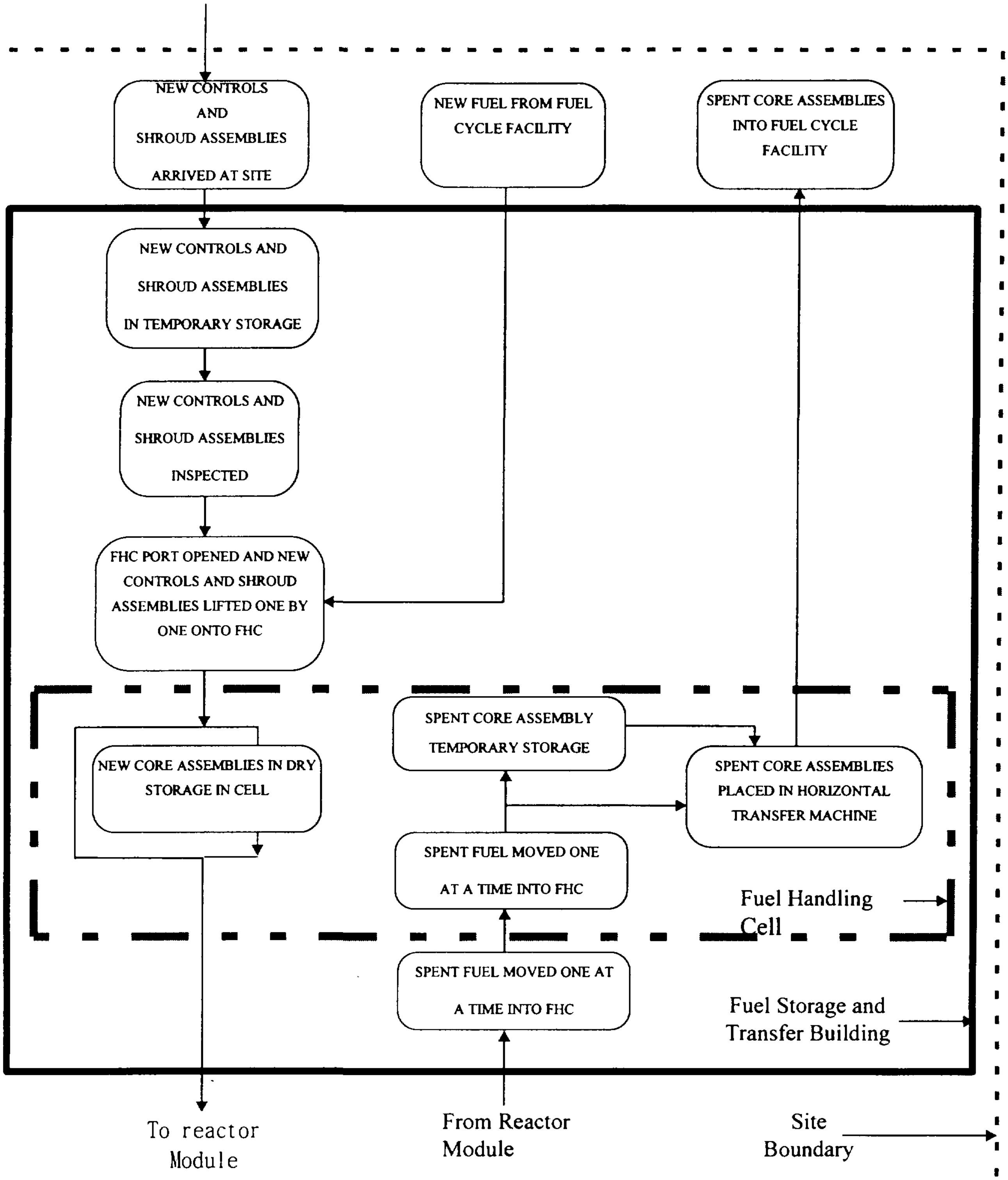


Figure 15 Fuel Flow Diagram

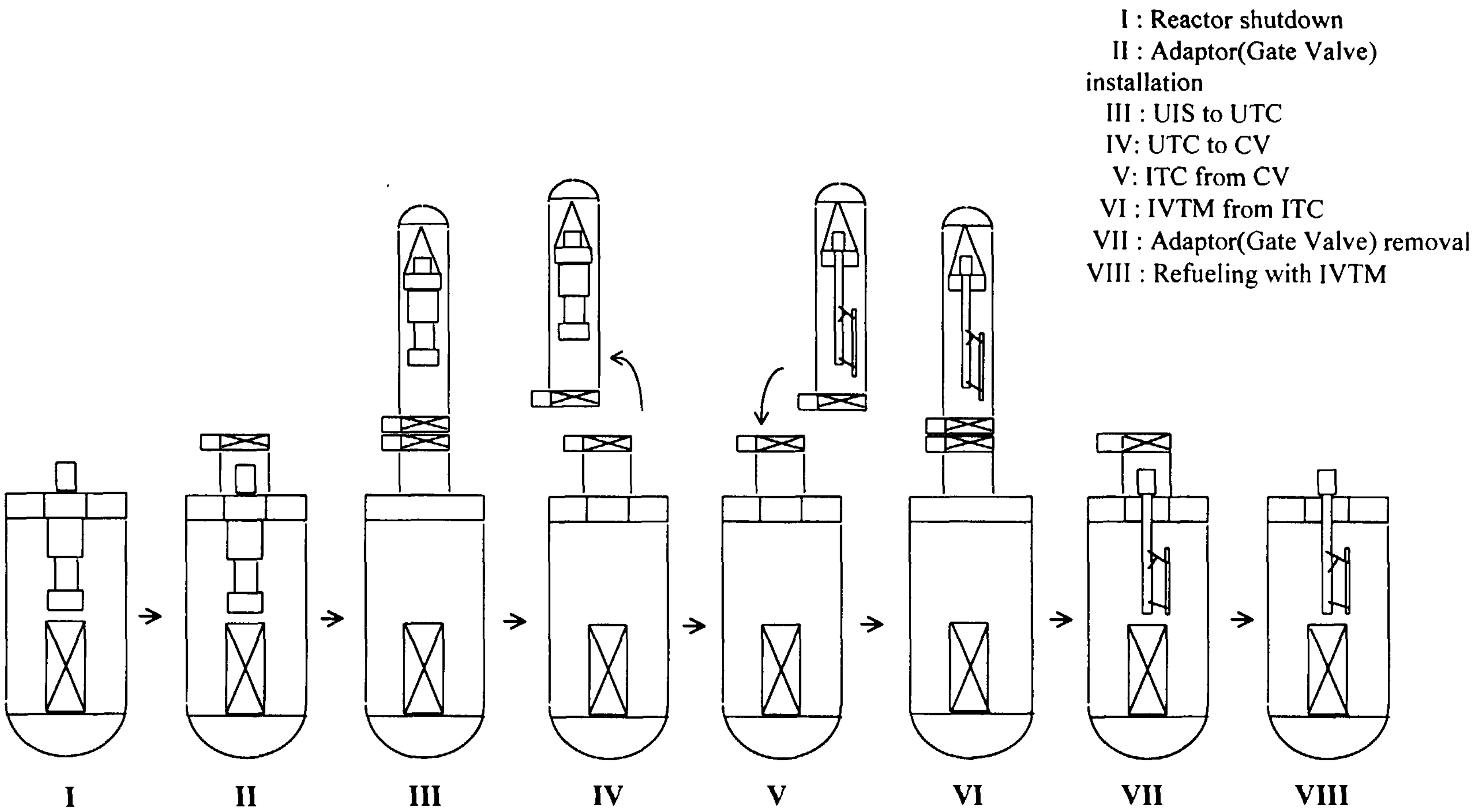


Figure 16 Conceptual Diagram of Plug-out/plug-in of the UIS System and IVTM Assembly

The new core assemblies are installed six at a time into a fuel transfer cask and transported to a module to refuel the reactor core. At the reactor module, new assemblies are exchanged for spent assemblies. Spent fuel assemblies are exchanged for previously stored month old spent fuel assemblies. The fuel transfer cask with 24 month old spent fuel assemblies is transported to the FSTB for temporary storage in the FTC or transfer to the co-located fuel cycle facility.

Important fuel handling facilities and equipments located in the FSTB include the FHC, new and spent fuel transfer facilities, storage facilities, and a communication center from which refueling and other fuel handling operations are coordinated. The fuel transfer cask, cask transporter are part of the transport system(TS) used to move core assemblies between the RSB and each reactor module during a refueling outage. The IVTM assembly(the IVTM and fixed plug) is located at the reactor module and is part of the reactor fuel handling system.

Prior to the start of refueling, a number of preparatory operations are conducted so that the reactor shutdown time is minimized.

All equipments and facilities to be used in the refueling and fuel handling operations are functionally checked out. This includes the communication center. Computers with associated hardware and software are checked out on simulators prior to use with the equipment.

The sequence for handling fuel begins with receipt of new fuel from the co-located fuel cycle facility. The new fuel is temporarily stored in a dry storage rack in the FHC, or the new fuel is moved directly into the FTC for transport to a reactor module for refueling. The FTC is a sealed and shielded structure that can carry six core assemblies.

30 days have been allotted for an average reactor refueling. This begins with reduction of reactor power from 100% to the power level from which the reactor is shut down The sodium in the reactor vessel is cooled down to a refueling temperature of 205 °C. The gate valve and adapter are installed over the reactor closure head. The UIS system, which consists of the UIS assembly and fixed plug, and the control rod drive mechanism are raised into the UTC and the UTC is stored in the casks vault(CV) within the reactor head access area. Then the ITC containing the IVTM assembly is transported into the top of the reactor closure head, the IVTM assembly is plugged into the reactor closure head, and the

above closure IVTM drive mechanism is installed. Concurrently, the reactor cover gas is purged and purified to reduce radioactivity levels in the gas to a very low level. The refueling port gate valve and adapter are installed and the gate valve closed. The FTC port plug is removed and temporarily stored in CV. This completes the preparation for refueling and permits operations to begin.

While final preparations are being made at the reactor, the FTC is loaded with six new fuel assemblies and transported from the RSB to the reactor module. The FTC is transferred into the reactor head access area and lowered into position onto the reactor gate valve port adapter. The IVTM commences the repetitive refueling cycle by removing a spent fuel assembly from the core and placing the assembly in an empty in-vessel storage position for 24 months storage. The refueling cycle includes picking up a new fuel element from the in-vessel transfer position, that was placed there by lowering one of the new core assemblies from the FTC. The new core assembly is transferred and installed into the vacated core position. The IVTM translates to an adjacent spent fuel assembly that is then removed and transferred to the empty in-vessel storage position. The IVTM translates to an adjacent stored 24 month old spent fuel assembly for removal into the emptied transfer position. The old spent assembly is raised into the FTC. The internals of the FTC are rotated to locate the next new fuel assembly in position over the reactor port and the new fuel assembly is lowered into the reactor transfer position for the IVTM to start replacement of the next core assembly. This cycle continues until one third of total core elements are exchanged. The TBD assemblies of blankets, shields and control are exchanged in a similar manner except not placed in storage for the 24 month period. At this point the reactor refueling of the TBD core assemblies is completed.

After refueling is completed, the IVTM assembly is raised into the ITC through the adaptor gate valve which is pre-installed for sealing, and the ITC is moved to the CV. Then the UTC is returned from CV to the reactor closure head and mounted on the adaptor gate valve, and the UIS system is plugged in the reactor closure head. And last the UTC and the adaptor gate valve are removed. The RFHS operations terminating refueling begin and are essentially the reverse of the preparation operations. The reactor instrumentation is checked out. The FTC port is sealed, and the reactor is made critical and returned to full power.

Since the spent fuel has decayed for TBD month in the reactor, the low decay heat fuel will be temporarily stored in the FRSSS or directly transferred into the co-located fuel cycle facility. Spent control, radial shield, CEM, and blanket assemblies are handled in the same manner as fuel assemblies except they do not require in-vessel storage for TBD months. These components can be shipped directly to the co-located reprocessing facility during refueling operation.

3.1.2 In-Vessel Transfer Machine (IVTM)

The IVTM assembly comprises the IVTM and a fixed plug. The fixed plug is similar to the one used for UIS but designed more simply since the reactor was shut down thus the buffer seal is not necessary and only a few bolts are sufficient to support the IVTM.

The IVTM is used to handle fuel assemblies, control rods, and other reactor components in the sodium-filled core of the reactor. The design is a modified pantograph machine with rotary seals. The machine is used only during reactor shutdown and is located in a penetration in the fixed plug. See Figure 17.

The machine is designed in two parts, the junction between the two parts is 1.5 m above the fixed plug. The drive section upper part is basically an electrically driven gear box for operating the in-vessel section lower part. It contains an electric motor, speed reducers, gears, torque limiting clutches, emergency hand operators and other components necessary to provide control and instrumentation.

The in-vessel section lower part is positioned vertically from the fixed plug and extends 7.5 m into the reactor. The machine is positioned at the center of the fixed plug. The machine can be rotated and the pickup leg driven outwards to position the grapple over the required fuel assembly and in-vessel fuel storage. The pickup leg can be moved radially outward to 1.8 m to position the grapple. The grapple can thus be positioned over any core assembly position, in-vessel fuel storage position and the fuel transfer position.

The IVTM can be rotated 360° on the centerline toward the center of the core. To position over the core center assembly, a detailed design study should be required later. Position accuracy of the machine is essential and is provided by the instrumentation

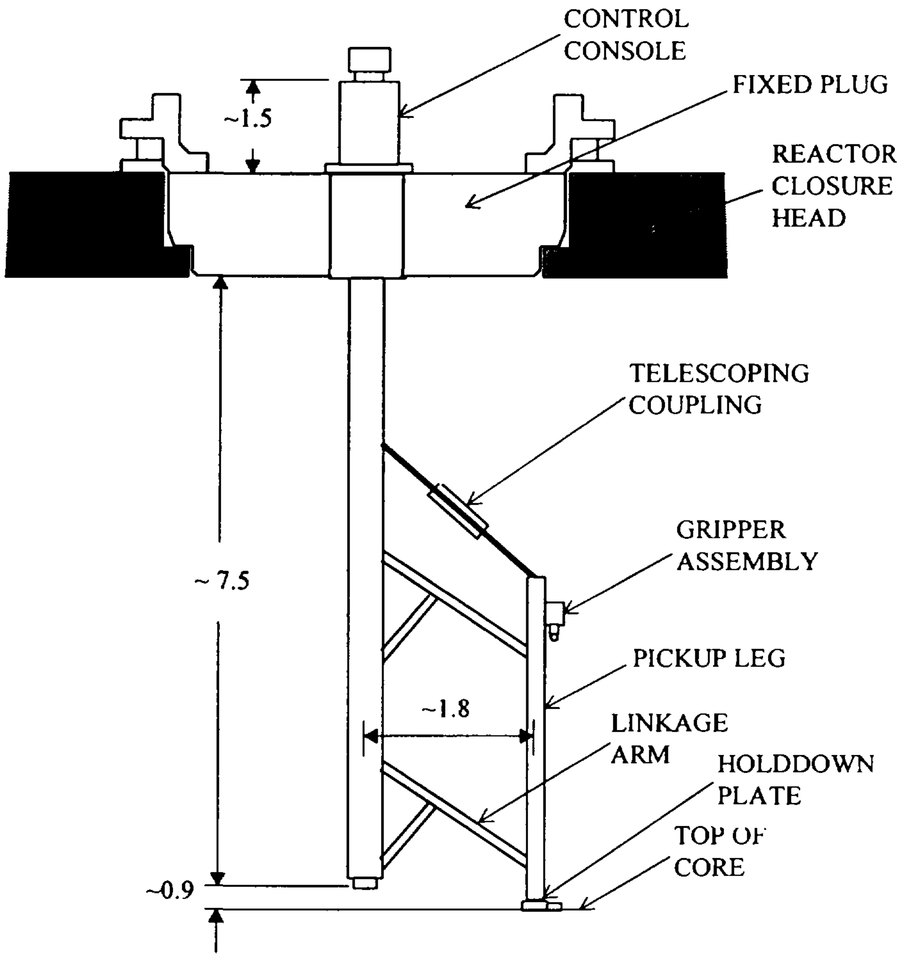


Figure 17 In-Vessel Fuel Transfer Machine (to be updated)

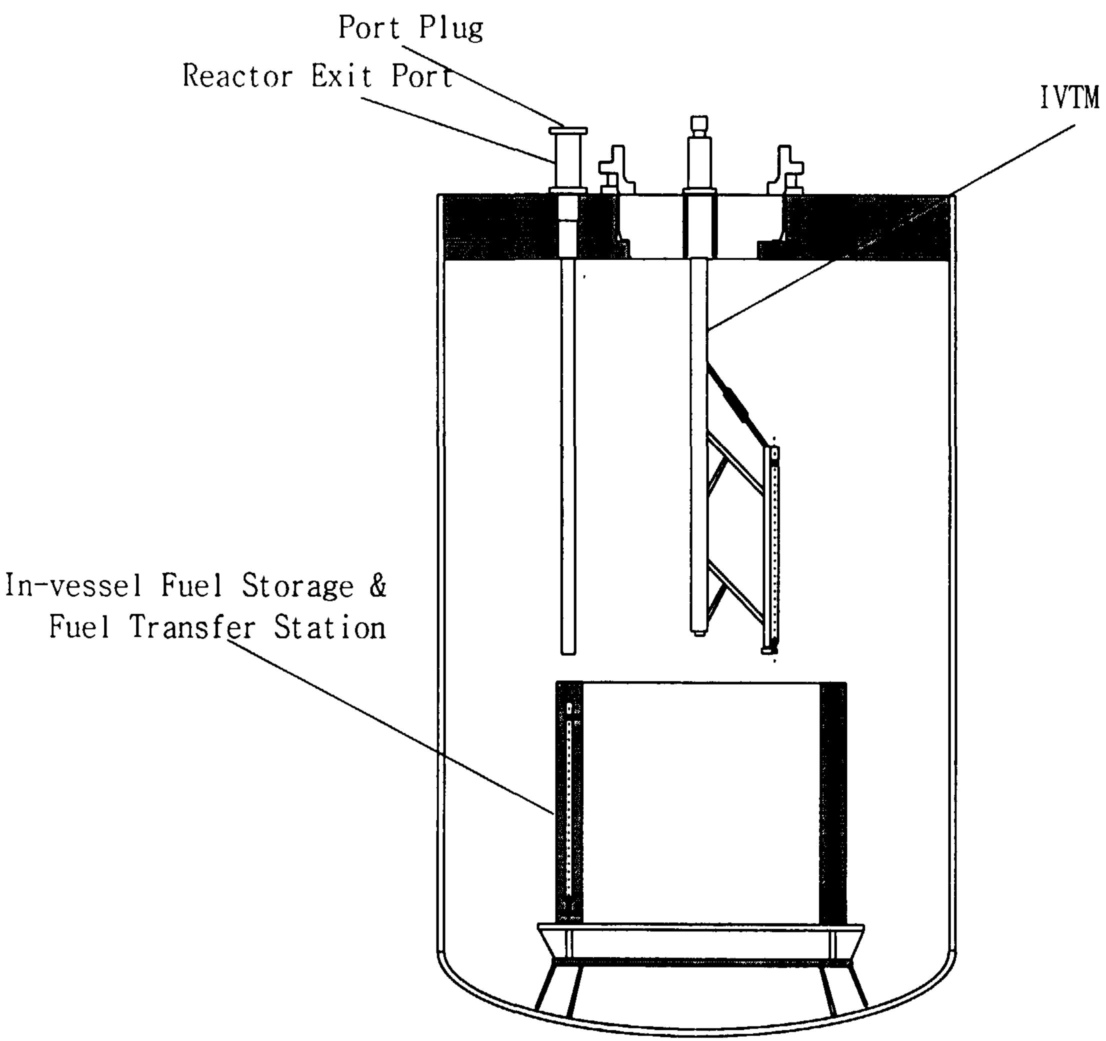
system. The instrumentation system will provide continuous position indication for the station control room of all machine movements.

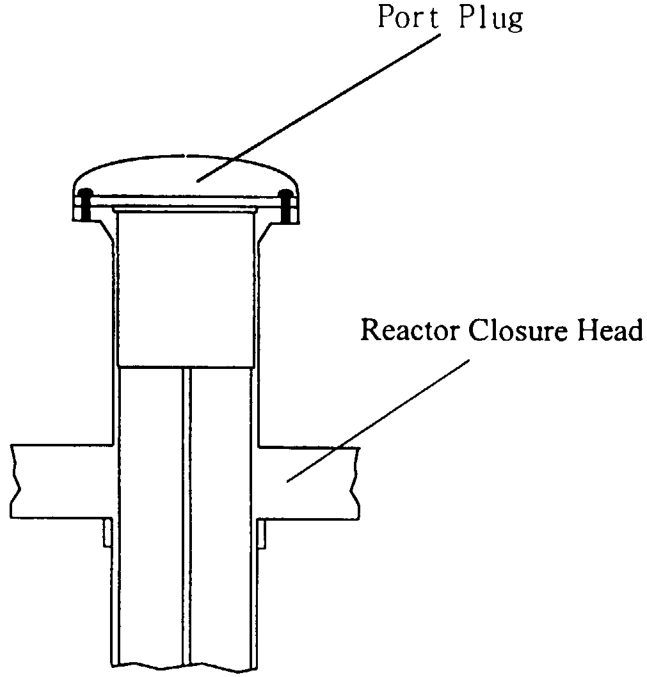
3.1.3 Reactor Exit Port and Plug

The reactor exit port in the reactor closure is fixed to the deck as shown in Figure 18. For refueling, an adapter with a gate valve will be mounted on the port onto which is mated the FTC (with gate valve) for the exchange of core assemblies(see Figure 14). The port is not cooled since the thermal analysis for the PRISM case indicated cooling of the port was not required.

The port is plugged and sealed during reactor operation. For refueling, this plug is unfastened and hoisted into a cask by the plug hoist and the cask moved to the CV for temporary storage. The plug hoist is driven from outside of the cask.

The refueling port plug consists of thermal and nuclear shielding. The lower end of the plug contains reflective shielding plates to match the design of the reactor closure. The plug closely fits the reactor port. The upper end of the plug is attached to the port cover, which is attached to the plug bail for lifting and removal. Following refueling, the refueling port plug is returned to the module, lowered into place and sealed.





Port Plug Details

Figure 18 Fuel Transfer Port & Fuel Transfer Station

3.1.4 UIS System Transfer Cask(UTC)

During refueling the UIS is removed and replaced with the IVTM assembly. Removal of the UIS system involves use of an adaptor with gate valve and the UIS transfer cask(UTC).

The purpose of the transfer adaptor is to provide a protected path between the opening in the reactor deck and the UTC. The adaptor consists of a cylinder with a gate valve on top and flange on the bottom as shown in Figure 19. The adapter is an extension of the primary system during the UIS removal and IVTM installation. The gate valve is closed when UTC is not coupled to the adaptor. The adaptor is used only during refueling and is moved from reactor to reactor as needed.

The UTC is illustrated in Figure 20. It is a tall cylindrical structure with a gate valve at the bottom and hoisting mechanism at the top. The cask inner diameter is 3.0 m and sized to accommodate the UIS and fixed plug. Some activation of the UIS structure near the core is expected thus shielding is used on the lower bottom of the cask cylinder.

At the top of the cask is a moving piston. The overhead crane and moving piston are used for raising or lowering the UIS in the cask. The moving piston is sealed to the cask cylinder with inflatable seals using Argon gas. The top of the piston is attached to the building overhead crane and the bottom of the piston to the UIS. An Argon gas control system is used to maintain an argon atmosphere within the cask and pressurize the moving piston inflatable seals. For removal or installation of the UIS, the UTC is mounted on the adaptor. The gate valves in the adaptor and cask are opened and the UIS is either lowered into or raised over the reactor. Once the UIS has been raised into the cask, the gate valve are closed allowing the cask with the UIS to be moved to the CV which is a temporary storage position within the reactor building.

3.1.5 IVTM Transfer Cask(ITC)

During refueling and after the UIS is removed, the IVTM assembly is installed in the reactor. The operation makes use of the same adaptor as for the UTC and ITC.

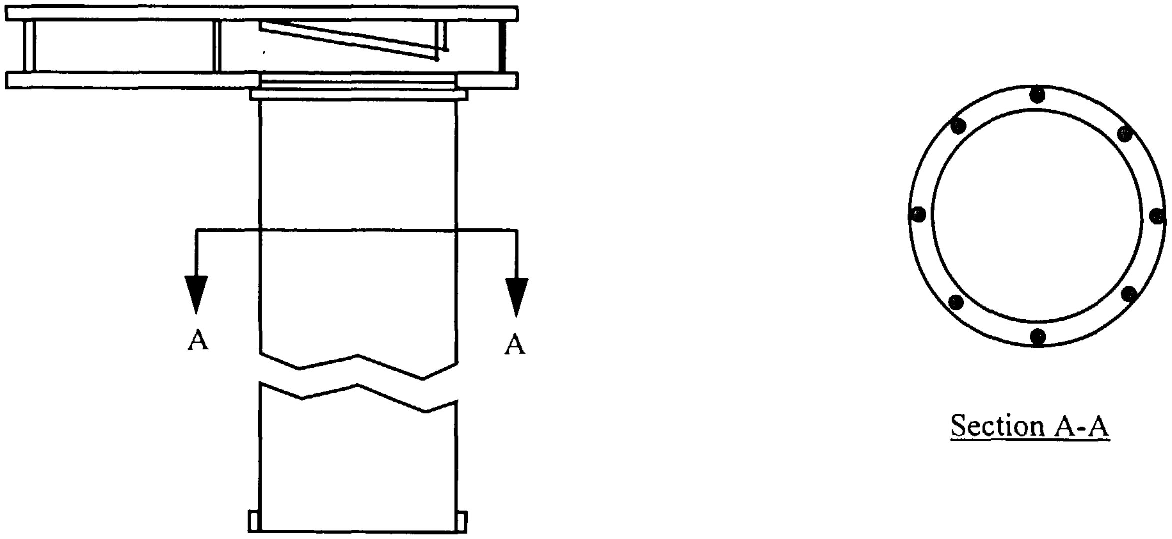


Figure 19 Cask Adapter

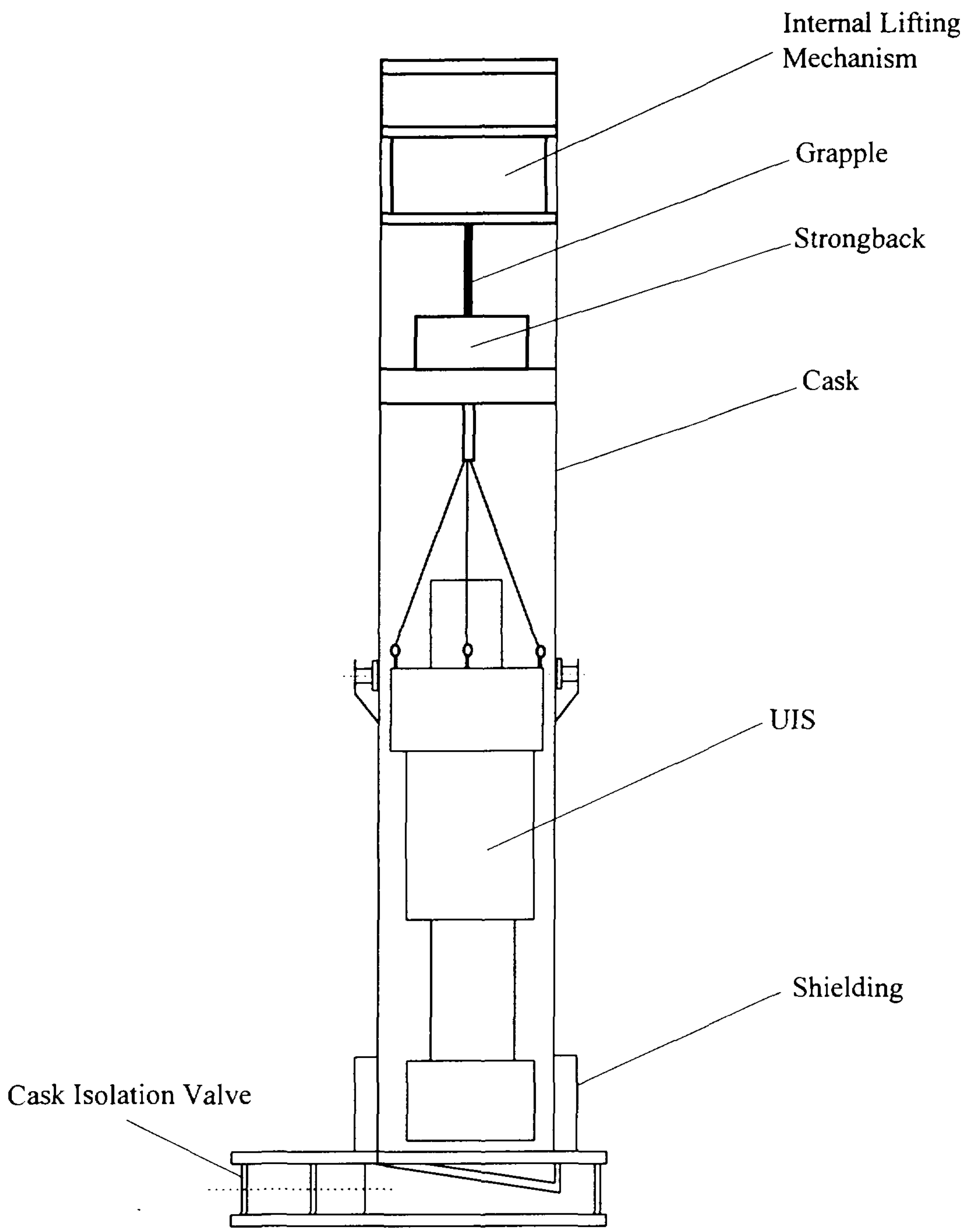


Figure 20 UIS Cask(UTC)

The ITC is illustrated in Figure 21 and is very similar to the UTC. It is a tall cylindrical structure with a gate valve at the bottom and hoisting mechanism at the top. The cask inner diameter is 3.0 m and sized to accommodate the IVTM assembly and the fixed plug. At the top of the cask is a moving piston. The overhead crane and moving piston are used for raising or lowering the IVTM assembly in the cask. The moving piston is sealed to the cask cylinder with inflatable seals using Argon gas. The top of the piston is attached to the building overhead crane and the bottom of the piston to the IVTM assembly. An Argon gas control system is used to maintain an Argon atmosphere within the cask and pressurize the moving piston inflatable seals.

For removal or installation of the IVTM assembly, the cask is mounted on the adapter. The gate valves in the adapter and cask are opened and the IVTM assembly is either lowered into or raised over the reactor. Once the IVTM assembly has been raised into the cask, the gate valve are closed allowing the cask to be moved to the CV which is a temporary storage position within the reactor building.

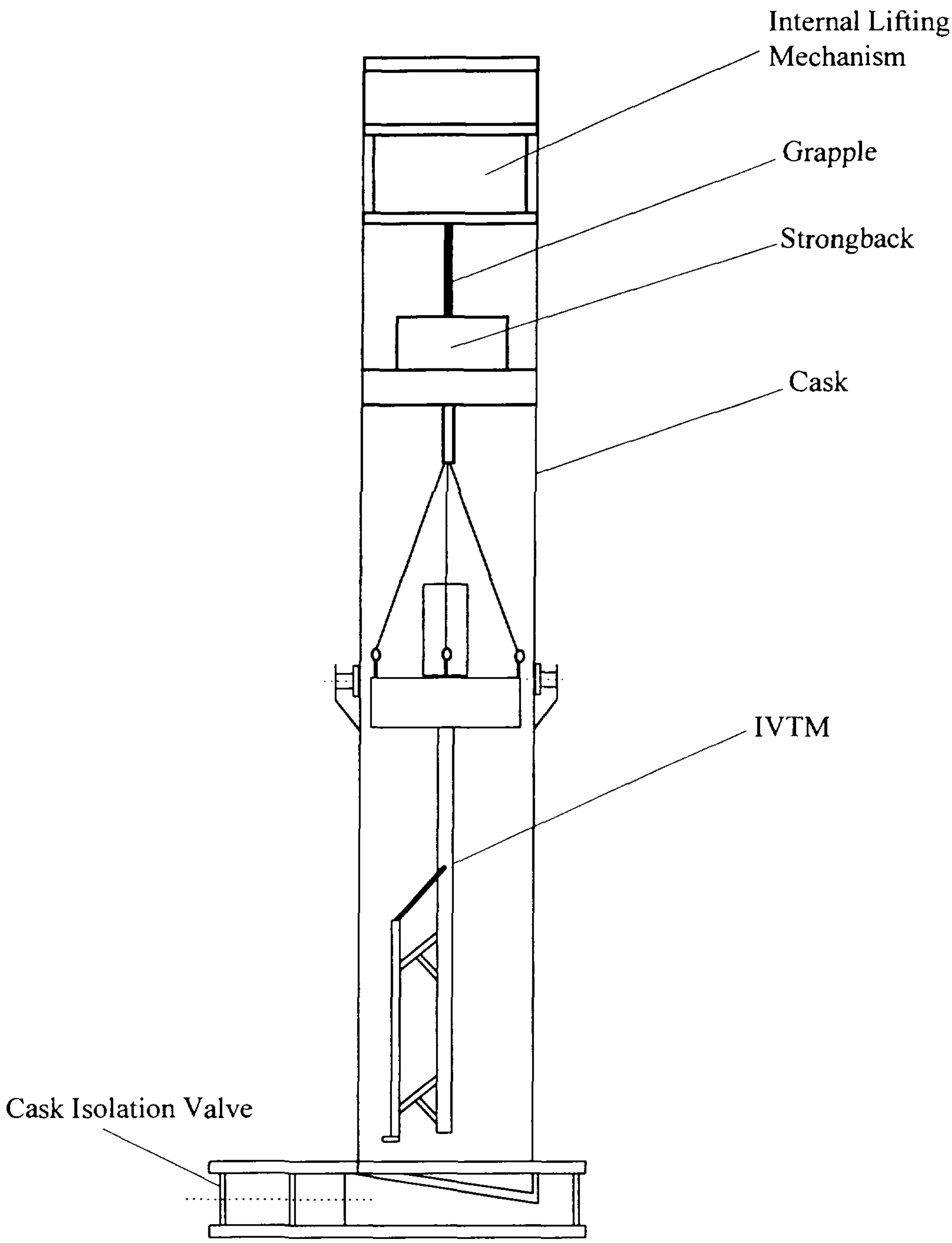


Figure 21 IVTM Cask (ITC)

3.1.6 Fuel Transfer Cask (FTC)

The FTC is a multi-element cask used to transfer six fuel assemblies or two(TBD) blanket assemblies between the FSTB and reactor module. The cask shown in Figure 22 is a shielded and inerted cylinder, 7.0 m high and 1.4 m outside diameter. The structure has a 39 cm composite wall structure consisting of concentric steel, depleted uranium, B_4C in a copper matrix (75% Cu) cylinders.

Within the cavity is a six-location carousel that is suspended, rotated, and positioned from the top. The carousel is motor driven to position each of the locations over the gate valve opening when desired to discharge an assembly from the cask. To discharge a core assembly, a drive deploys a bi-stem that is attached to the upper inside of the pot. The stem stays attached to the pot while in the reactor building but is able to be detached for maintenance at the FSTB. By locating the bi-stem at a specific location on the pot, the IVTM has the freedom to move into position over the pot, engaging and removing the core assembly for transfer within the core area and thus reduce the number of steps during a refueling cycle.

3.1.7 Cask Transporter(CT)

The cask transporter shown, in Figure 23, is a mobile, self-propelled rail transporter that moves a FTC. The FTC is raised and lowered to mate with the fuel transfer ports by a hydraulically operated system on the cask transporter(CT). An alignment mechanism allows the operator to position the transporter above the reactor being refueled and the RSB fuel transfer port. The cask transporter is equipped with two pressure storage tanks and a compressor for purging the gas lock area between the FTC and the reactor fuel transfer port. One tank contains clean helium and the other, shielded, contains purged helium. The cask transporter is powered by a diesel engine and is equipped with an enclosed cab for the operator. The cask transporter is constructed of structural steel and is classified non-seismic Category I, Type A. The cask, however, is classified Category I and is tornado hardened.

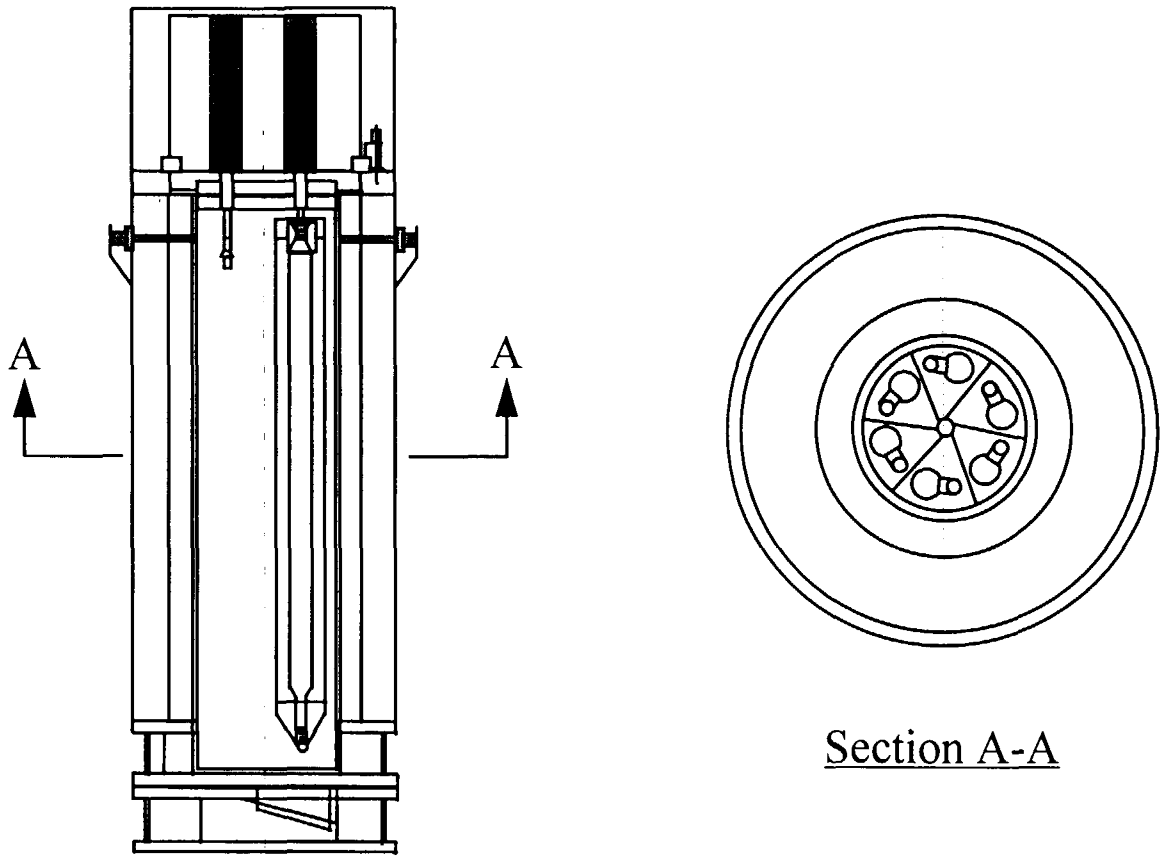


Figure 22 Fuel Transfer Cask

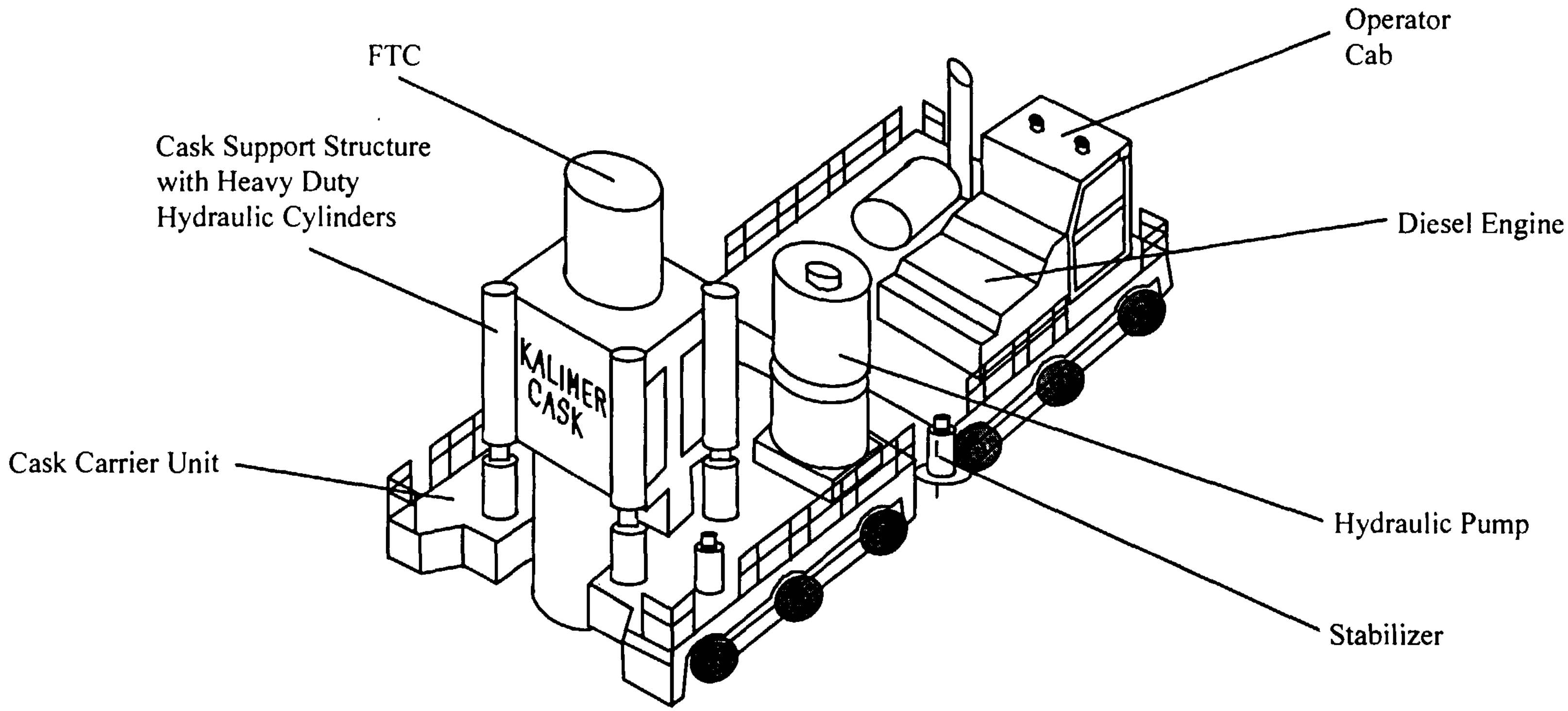


Figure 23 Cask Transporter

3.1.8 Fuel Handling Cell and Fuel Shipping

The fuel handling cell, shown in Figure 13 consists of the cell structure which provides the inert gas containment, shielding, access, viewing, and transfer ports between the FTC and the co-located fuel cycle facility. The telescoping tube hoists and transfer carts are used to support and transport the fuel assemblies between transfer stations, storage racks and ports in the cell.

3.2 Design Bases Requirements and Accommodation Approaches

3.2.1 General Functional Requirements

The reactor refueling system (RRS) provides the services for handling all core assemblies and related equipments at the reactor site. The primary functions of the Reactor Refueling System are as follows:

1. Receive, inspect, store and prepare new core assemblies for insertion into the reactor.
2. Transfer assemblies between facilities (e.g., reactor and reactor service building).
3. Transfer core assemblies between the core and in-vessel storage or transfer positions.
4. Provide temporary storage for spent core assemblies before transfer to the off-site reprocessor facility.
5. Prepare spent core assemblies for shipment to off-site reprocessor.
6. Provide inventory control of all core assemblies. Core assemblies are defined as fuel. blanket, control, and radial shield assemblies.
7. Removal and insertion of the UIS and IVTM.

Accommodation Approach:

The RRS facilities and equipment described in the previous section will satisfy the functions specified above. The RRS concept is illustrated in Figure 13~ 14 The above functions are satisfied as follows:

- The fuel storage and transfer building contains the new fuel storage cells and equipment for receiving, inspecting, storing and preparing new core assemblies for insertion into the reactor.

- A cask and cask transporter are provided for transferring assemblies between facilities the reactors and reactor service building.
- A plug-in type IVTM is provided for transfer of core assemblies between the core and in-vessel storage and transfer positions.
- Storage racks in the fuel storage and transfer building are provided for temporary storage for spent core assemblies before transfer to the off-site reprocessor facility.
- The fuel storage and transfer building also contain the equipment for preparing the spent core assemblies for shipment to off-site reprocessing.
- An inventory control system will be established later in the design process to provide inventory control of all core assemblies.
- Casks, adapter and hoisting equipment are provided for removal and inserting the UIS and IVTM for each refueling outage.

3.2.2 Design Requirements

General Requirements

The RRS consists of the systems and components required to receive, replace, store, and ship the core assemblies. These functions are accomplished by the reactor fuel handling system (RFHS), transport system (TS), and the fuel receiving storage and shipping system (FRSSS). The requirements are as follows:

1. The RRS design and performance parameters are as follows:

   Fuel Handling:

   1) Interval

      - 24 months

   2) Spent Fuel Storage Mode

      - In-Vessel Storage

   3) Reactor Preparation

      - UIS system removal and installation of the fuel handling machine assembly

4) Refueling Temperature (Shutdown)

- 205 °C

Accommodation Approach:

The RRS concept is based on a 24 month refueling interval, in-vessel under-sodium storage and use of a plug-in IVTM. The design of the plug-in IVTM will be compatible with in-sodium use.

2. The RRS shall be designed for a 60-year life. Systems and components that cannot achieve a 60-year lifetime shall be designed to satisfy the requirement by means of maintenance, replacement, or redundancy; and shall not impact on the plant availability requirement.

Accommodation Approach:

The RRS facilities and equipment will be designed to function for 60 years. For example, the plug-in IVTM will be made from stainless steel, Inconel and other special material compatible with long-term operation under sodium. Elastomer seals, bearings, motors and other hardware which has a limited life or a potential for failure will be designed to be replaced. The other RRS equipment will satisfy the 60 year requirement in similar fashion.

3. The safety related systems (SRS) for the RRS shall be designed to be operable during an operating basis earthquake (OBE). The SRS shall be able to be safely shut down during a safe shutdown earthquake (SSE), and be capable of being maintained in a safe shutdown condition.

Accommodation Approach: See structural evaluation

4. The minimum design refueling interval shall be 24 months. All refueling, planned maintenance, and inspections requiring plant shutdown shall be accomplished during the planned, TBD day shutdown period every twenty four months.

Accommodation Approach: Detailed approach is determined later(in conceptual design stage)

5. Maintenance and inspection shall be an integral function of the design process. Provisions for inspection, maintenance, and removal/replacement, including contingency for abnormal maintenance activities, shall be provided.

Accommodation Approach: To be designed

6. The design shall provide protection against design basis accidents.

Accommodation Approach: See structural evaluation

7. Spent fuel radiation levels and decay heat shall be based on a five-year irradiation life source term for the appropriate assembly type.

Accommodation Approach: See thermal analysis of FTC.

Reactor Fuel Handling System

The reactor fuel handling system (RFHS) shall satisfy the following requirements:

1. The RFHS shall be capable of replacing the core components, including the fuel, blanket, radial shield, CEM, and control assemblies.

Accommodation Approach:

The IVTM, fuel transfer cask, in-vessel storage ranks and ex-vessel storage ranks are designed to accommodate the 6 types of core assemblies that is the fuel, blanket, radial shield, CEM, and control assemblies. Each of these core assemblies has a common top handling socket for grappling the IVTM grappling head and the fuel transfer cask grappling head. The external features of the core assemblies are essentially identical for fitup with the in-vessel fuel storage ranks, fuel transfer cask bucket and ex-vessel storage ranks

2. Refueling shall be accomplished with the primary coolant at 205 °C and the reactor cover gas at atmospheric pressure.

Accommodation Approach: This parameter affects the IVTM and the fuel cask transfer bucket. For these, materials will be selected that are compatible with in-sodium operation.

The transfer bucket will be made from stainless steel and the IVTM from stainless steel, Inconel and other higher alloys.

3. The RFHS shall be capable of starting refueling 4 days after reactor shutdown and completing refueling within 30 days after reactor shutdown.

Accommodation Approach:

Task analysis and time/motion studies will be performed to show that preparation for refueling and the refueling itself can be completed within the specified times. A preliminary evaluation of the tasks and time for repairing for refueling is given below:

Task Description	Allotted time
Reactor shutdown and cool to refueling temperature	1~2 days
UIS assembly removal • Installation of UIS cask adapter • Installation of UIS cask • Raising UIS assembling into cask • Cask/UIS assembly removal	~ 2 days
IVTM installation • Installation of IVTM cask adapter • Installation of cask/IVTM • Insertion IVTM into reactor • Cask removal • Attachment of plug to reactor head	~2 days
Fuel transfer cask installation • Installation of fuel transfer cask adapter • Removal of transfer port plug • Position and attachment of fuel transfer cask.	~2 days
Various reactor and refueling system checks	~1 day
Total duration	9 days

4. Spent fuel requiring cooling in excess of the RFHS cooling capability shall be stored within the reactor under sodium for a period of 24 months.

Accommodation Approach:

The reactor internals are fitted with fuel storage racks. The racks are located external to the core barrel and are arranged in a circle. The drive duel and blanket assemblies will require in-vessel storage under sodium to allow the decay heat level to reach about 1.5 kW so as to be within the fuel transfer cask passive cooling and the ex-vessel storage cell passive cooling. The control assemblies, GEMs and shield assemblies do not require in-vessel storage and thus can be removed from the reactor at the end of life.

5. Failed fuel assemblies will require no special handling. Failed fuel assemblies will be stored in the reactor for one cycle before transfer to the fuel cycle facility.

Accommodation Approach: To be considered

6. A plug-in In-Vessel Transfer Machine(IVTM) shall be used to transport the fuel under the head within the reactor vessel.

Accommodation Approach: The IVTM design concept illustrated in Figure 17 will be designed to transfer the core assemblies within the reactor. The radial and axial motions from the IVTM rotation, pantograph extension and gripper travel will allow it excess all core positions, in-vessel storage positions and the fuel transfer position.

7. A fixed fuel transfer position shall be used to transfer fuel between the reactor vessel and the fuel transfer cask (FTC).

Accommodation Approach: The reactor design included a fixed fuel transfer position as shown in Figure 14. This consists of the tube that extends from the fuel transfer port to near the bottom of the vessel. This tube guides the transfer bucket lowered from the cask into the reactor. The has a long cut-away to allow the IVTM to access the top of the transfer bucket to remove new core assembly and insert a spent assembly.

8. The reactor fuel transfer port shall be located in the fixed deck.

Accommodation Approach: As shown in Figure 18, the fuel transfer port is located in the fixed deck.

9. The control system for each RFHS component shall be designed as an integral segment of the overall reactor refueling control system. All RFHS components shall be capable of being activated by manual control as a backup mode.

Accommodation Approach: To be designed

10. The IVTM grapple shall be able to rotate core assemblies for alignment into core position.

Accommodation Approach:

The IVTM design shown in Figure 17 has a gripper assembly mounted on the pickup leg. The gripper assembly will be designed with a rotation capability for azimuth alignment of the assembly with the core position.

11. The IVTM holddown plate shall hold down adjacent core assemblies during withdrawal for the full length of the fuel assembly.

Accommodation Approach:

As illustrated in Figure 17, the IVTM design includes a hold-down plate located at the base of the pickup leg. This plate covers the tops of the adjacent 6 assemblies to prevent their withdrawal during removal of the grappled assemblies. The US FFTF IVTM has successfully employed a hold-down plate the keep the adjacent assemblies in-place.

Transport System

The transport system (TS) consists of the fuel transfer cask (FTC), the UIS transfer cask (UTC), IVTM transfer cask (ITC), and the cask transporter (CT). The UIS system consists of the UIS, the control rod drive mechanism(CRDM), and a fixed plug. The IVTM assembly is comprised of the IVTM and a fixed plug. The TS shall satisfy the following requirements.

1. The TS system shall provide a means of handling and transferring the FTC with new or spent core assemblies between the reactor and the fuel cycle facility. The transfer shall not contaminate the reactor head during normal refueling.

Accommodation Approach:

The fuel transfer system for transferring spent fuel from the reactors to the fuel handling cell is illustrated in Figures 13 and 23. Connection of the fuel transfer cask to the reactor is with an adapter. This unit interfaces with the reactor head and the fuel transfer cask. The adapter gate valve and that of the cask allow extension of the reactor seal cover gas boundary to prevent air contamination into the reactor and the exposure of the sodium wetted fuel. When the cask is not in-place, the closed gate valve on the adapter maintains the sealed reactor cover boundary. An on-line cover gas cleanup system will maintain cover gas purity levels during fuel transfer.

2. The CT shall provide a method of supporting and mating the FTC, UTC, and the gate valves at the RSB and reactor module.

Accommodation Approach:

The cask transporter, fuel transfer cask and adapter arrangement are illustrated in Figure 14. The cask is supported by the reactor building floor. The adapter connects and seals the cask to the reactor. The same connection and sealing arrangements are employed at the reactor service building.

3. The TS shall provide a means of handling the caissons for replacing:

a. Control rod drive assemblies

b. UIS

c. IVTM(refueling machine)

d. Intermediate heat exchanger

e. EM pump

f. Miscellaneous components and special tools

Accommodation Approach:

The cask transport arrangement consisting of the cask, cask carrier and locomotive is shown in Figure 23. This unit is mounted on steel rails connecting the reactor buildings and the reactor service building. A similar transport arrangement will be employed for the UIS, IVTM, intermediate heat exchanger, EM pump, and miscellaneous components.

4. The FTC shall provide locations for handling six core assemblies in the sodium-wetted drip condition.

Accommodation Approach:

The FTC will be designed so that it may accomodate six core assemblies with a six-location carousel that is suspended, rotated, and positioned from the top. The carosul is motor driven to position each of the locations over the gate valve opening. The drip pan is also installed over the gate valve to store the sodium dripped from the fuel assemblies.

5. The transporter shall be capable of freely moving between the RSB and any reactor module while transferring a FTC with core assemblies - or caisson with reactor components.

Accommodation Approach:

The rail road system installed on the co-located ground level will make the transporter move freely between the RSB and any reactor module as shown in Figure 13.

Fuel Receiving, Storage, and Shipping System (FRSSS)

The FRSSS consists of the fuel handling cell, for receiving and temporary storage of new and spent core assemblies, and the integrated refueling control system which controls portions of the FRSSS in conjunction with the FHS. The FRSSS shall satisfy the following requirements:

1. The FRSSS shall be located within the Reactor Service Building.

Accommodation approach: To be designed

2. Storage shall only be provided to accommodate one refueling change-out load.

Accommodation approach: To be designed and considered

3. New fuel shall be dry stored in the RSB. Prior to transfer to and insertion into the reactor, the assemblies will be preheated.

Accommodation approach: To be designed.

4. The design shall assume that a co-located fuel cycle facility will be available on site for processing metal fuel(not for oxide fuel). Additionally, the design shall be capable, with minimal modifications, of providing for off-site shipment of spent assemblies.

Accommodation approach: To be designed and considered.

5. The FRSSS shall be capable of transferring all the spent fuel from one refueling to an on-site co-located facility and to accept a reload of new assemblies in the period before the next scheduled refueling.

Accommodation approach: To be designed and/or considered.

6. The FRSSS design shall preclude attaining $K_{eff} \geq 0.95$(K_{eff}: criticality) during both normal and malfunction conditions.

Accommodation Approach:

Criticality safety analyses should be performed to assure that the FRSSS is subcritical during both normal and accident conditions. These analyses include criticality evaluation of fresh fuel receiving, fresh fuel storage in the RSB, fresh fuel transfer to the reactor, spent fuel transfer from the reactor, and spent fuel storage. The criterion used for the criticality safety is to assure K-eff less than 0.95 during normal condition and malfunction conditions in which optimal neutron moderation is assumed.

7. The FRSSS shall be capable of receiving, storing, and transferring fuel handling operations while the plant is at full power.

Accommodation approach: To be designed

8. The FRSSS shall be capable of starting refueling four days after reactor shutdown and completing refueling within 30 days after reactor shutdown.

Accommodation approach: Same as RFHS case

9. Inventory monitoring and safeguard control for special nuclear materials shall be provided at all times and locations at the reactor site.

Accommodation approach: To be designed

10. Control of the refueling operations shall be provided. Information and data shall be provided to the Plant Control System for monitoring of the refueling operations in the control center.

Accommodation approach: To be designed

New Fuel Receiving

The plant fuel handling system shall be designed to receive new core assemblies from either truck or rail shipment.

Accommodation approach: To be designed

Spent Fuel Shipping

The plant shall be designed to transfer spent fuel to a co-located fuel cycle facility for processing. Additionally, the design shall be capable of providing for off-site shipment of spent assemblies.

Accommodation approach: To be designed

3.2.3 Process Requirements

Components Handled

Handling core components shall be based on core assembly configurations and data as follows:

1. Design of the RRS shall be based on handling new, spent, and failed core assemblies.

Accommodation approach: To be designed

2. Design of the RRS shall be based on all core components handled by the transfer machines having identical handling sockets at their upper end and identical lengths of the core assemblies.

Accommodation Approach: To be designed

3. The RRS shall provide the capability to handle components at a rate necessary to support refueling operations.

Accommodation Approach: To be designed

4. Heat transfer calculation shall be based on normal and greatest possible thermal loads. For example, the thermal effect on a fuel assembly stuck inside the inert atmosphere of the FTC shall be based on the fuel assembly with the greatest decay power after 24 months in-vessel storage or a blanket assembly after reactor shutdown. The effect in the hottest fuel from a normal refueling sequence shall also be determined.

Accommodation Approach:

The thermal analysis for FTC shall be performed to determine the heat rejection capabilities of the FTC. The FTC thermal analysis shall have the following procedure.

- The cross-sections of FTC and fuel assembly are modeled in more detail considering the Helium atmosphere inside the cask and thermal radiation gaps between all the material layers. The thermal radiation gap model is used to account for the contact resistance between materials.

- The boundary conditions used in the model involve convection, conduction and radiation modes of heat transfer. Adiabatic boundry conditions are generally assumed on each model along the lines cutting the 60-degree wedge. For radiation heat transfer the lines of symmetry is treated as diffuse surfaces for the simplicity of the calculation and conservative result. On the outside of the cask, a convection boundary condition is also used.

- A two-dimensional steady state thermal analysis of the steel, uranium, and boron carbide layers axially of the structures is firstly performed using finite difference algorithm since th finite element model could result in a smaller peak temperature. Then the detailed cross-section thermal analysis is implemented using both finite difference and element tools.

- The temperature distributions resulted from the two models(finite difference and element models) are analyzed and illustrated. Finally the peak and minimum temperatures of the FTC model are predicted.

5. Design of the RRS shall consider the types of handling operations and their frequencies given in Table 3.

Accommodation Approach: To be designed

In-Reactor Vessel Fuel Storage Capacity

The RFHS shall interface with Reactor System which provides in-vessel storage for normal core component handling operations.

Accommodation Approach:

The in-reactor vessel fuel storage is designed to store the spent fuel assemblies for an additional 24 months in wetted condition in order to sufficiently lower the decay heat levels of the assemblies.

Transfer Limitations

The transfer machine load and speed capabilities shall be limited as follows:

1. All RRS transfer machines shall be designed to limit the maximum push and pull forces they are capable of exerting to prevent damage to core assemblies, the transfer machines or other components under unusual handling operations.

Accommodation Approach:

The system dynamic and component analysis shall be performed to determine the limit forces for the transfer machines based on the structural requirements.

2. All RRS transfer machines shall be designed to limit push and pull forces during normal operations in order to apply only expected, reasonable handling loads on the core assemblies.

Accommodation Approach:

The system dynamic and component analysis shall be performed to determine the limit forces for the transfer machines based on the structural requirements.

Table 3 Frequency Of Normal And Infrequent Handling Operations

Operations	Number of Events During Life of Plant*	
	Expected	Design
Normal Operations		
Normal Refueling	29	40
Infrequent Operations		
Initial Reactor Loading	1	1
Complete Reactor Loading	0	1
Final Reactor Unloading	1	1

*60 year plant design life

3. All RRS transfer machines shall be designed to limit maximum handling speeds for new and spent core assemblies in order to prevent possible damage during normal handling.
Accommodation Approach: To be designed

4. All RRS transfer machines shall be designed to minimize the impact load when seating a core assembly, grapple, or bucket by limiting the maximum setdown speeds.
Accommodation Approach: To be designed

5. All RRS transfer machines shall have electrical or mechanical interlocks that prevent release during normal preparation of a grappled core assembly until the assembly is seated.
Accommodation Approach: To be designed

6. All RRS core assembly transfer machines shall include an overspeed control and fail-safe brake on the hoisting systems.
Accommodation Approach: To be designed

Misalignments

The following misalignment capabilities are necessary for handling core components because of normal assembly, thermal and operational tolerances.

1. RFHS equipment shall be designed with sufficient compliance and lead-in (combined with the lead-in of the component to be grappled) to permit grappling and insertion without excessive side loads, under total misalignments between the grapple and core component and between the core component and the core position, transfer position, or temporary storage position.
Accommodation Approach: To be designed through the tolerance and system dynamic analyses

2. The tolerances and alignments of equipment shall be limited, so that the installed and operational misalignments from true position do not exceed specified amounts.

Accommodation Approach: To be designed through the tolerance and system dynamic analyses

In-Vessel Transfer Operations

The in-vessel fuel handling operations are restricted as follows to prevent fuel motion that would cause inability of the IVTM to normally find and grapple a core assembly and to prevent damage to core upper structure, or other equipment.

1. The procedure used during a normal refueling shall be to replace one core assembly at a time, such that two open lattices adjacent to each other do not occur in the core.
Accommodation Approach: to be scheduled to reflect the procedure

2. The number of open lattice positions shall not be restricted for unusual operations using special core assemblies (e.g., complete reactor unloading).
Accommodation Approach: to be scheduled

Core Assembly Cooling

During transfer operations, the RRS shall be capable of cooling fuel assemblies for an indefinite period as follows:

1. The RRS shall provide, or other supporting system shall provide, cooling of core assemblies to limit their steady-state temperatures to the values specified in Table 4 with decay heat specified in Table 5.
Accommodation Approach:
The thermal analysis for each RRS equipment shall be done to prove the cooling capacity based on the procedure mentioned previously.

2. RRS equipment and facilities handling and storing spent fuel assemblies shall be provided with inherent means of cooling. Each means of cooling shall have the

Table 4 Irradiated Fuel Assembly Design Temperature During Fuel Handling And Storage

Operation	Peak Mid-Wall Cladding Temperature Limit(°C)
Normal Operation and Anticipated Events	
Handling	621 °C(TBU*)
Short-Term Storage	621 °C(TBU)
Unlikely and Extremely Unlikely Handling Events	677 °C(TBU)

*TBU: To be updated later

capability to remove the core assembly design decay heats specified in Table 5.

Accommodation Approach:

The thermal analysis for each RRS equipment shall be done to prove the cooling capacity based on the procedure mentioned previously.

Ex-Vessel Storage Capacity

A dry inert gas temporary storage area for 58(TBD) new core assemblies shall be provided.

Accommodation Approach: To be considered.

Table 5 Design Decay Heats For RRS

Equipment or Facility	Design Decay Heat (kW)
In-Reactor Storage	
Normal Refueling	20(TBU)
Hottest Fuel	35(TBU)
Fuel Transfer Cask	
Normal Refueling Load - 3 Fuel Assay's, 2 Blanket Assay's	3.9(TBU)
Fuel Handling Cell (FHC)	
Fuel Handling Machine	1.3(TBU)
New Fuel Receiving Equipment and Storage	
Total for Refueling - Prior to Transfer to Reactor	0(TBU)
Single Assembly	0(TBU)

Receiving of New Core Assemblies

a) General

All new core assemblies shall be received and stored during the year preceding the next reactor refueling as follows:

1. Immediately upon receipt from the co-located reprocessing facility, new fuel shall be placed in the storage racks of the fuel handling cell (FHC).

Accommodation Approach: It shall be accomplished by using FTC, CT and a fuel assembly transfer system in the FHC.

2. The new fuel storage facilities and handling equipment shall be designed to accommodate proliferation-resistant new fuel.

Accommodation Approach: To be designed separately

3. New core assemblies such as control rods, and radial shields shall have a "hands on" inspection facility located in the receiving area. The receiving area shall have storage space for incoming new core assembly containers.

Accommodation Approach: To be designed

4. The FRSSS design shall preclude contamination of new core assemblies which could lead to potential flow blockage.

Accommodation Approach: To be considered as a precaution for RRS

b) New Core Assembly Inspection

New fuel and blanket assemblies will receive all necessary inspections and verification at the fuel service facility prior to transfer to the FHC. All incoming new control and shield assemblies shall be inspected as follows:

1. The FRSSS shall be designed to examine shipping containers externally for evidence of damage.

Accommodation Approach: To be designed

2. The FRSSS shall be designed to perform standard inspection of new core assemblies as listed below.

a. Verify assembly identification by means of visual and mechanical examination of the assembly serial number and coding, respectively. Verify proper discriminator socket geometry, using "go/no go" gages.

b. Visually verify absence of dents, nicks, and gouges, especially in the areas of: (1) hexagonal load pad corners, (2) inlet nozzle, (3) piston rings, (4) discriminator socket, and handling socket interfaces.

c. Visually examine the internals of the outlet and the inlet nozzle to verify the absence of foreign objects or material.

d. Visually examine shipping shock indicators.

e. Visually examine core assembly duct for travel-induced bow or deformation.

f. Inspect control rods for free operation of neutron absorber column with friction force.

Accommodation Approach: To be designed

3. The FRSSS shall be designed to provide a non-standard inspection, as listed below, of core assemblies for which defects were observed during the standard inspection.

a. Photograph surface defect.

b. Perform selected dimensional inspection of external features(i.e., outside diameter or distance across hexagonal flats) at any axial location. This inspection shall be performed using general-purpose tools and gauges available commercially.

Accommodation Approach: To be designed

4. Unacceptable core assemblies shall be dispositioned as determined by applicable quality assurance procedures; however, no additional FRSSS equipment shall be provided for this purpose.

Accommodation Approach: To be considered in the quality assurance procedures

Transfer Operations

a) Transfer of UIS system

The following operations are used to transfer the UIS system into the CV within the reactor head access area.

1. Transfer operation within the reactor module shall take place only when the reactor is shut down.

Accommodation Approach: To be considered as a prerequisite before starting the transfer operation within the reactor module.

2. The transfer of UIS system into the CV within the reactor head access area shall be done with the UIS transfer cask(UTC).

Accommodation Approach:

The UTC shall be designed as shown in Figure 20 and the overhead crane shall be used or an alternative system shall be designed for transferring the UTC into the CV.

b) Transfer of the UIS and IVTM assemblies

The following operations are used to remove the UIS and install the IVTM assembly by transferring from the reactor closure head to the CV.

1. Transfer operation within the reactor head access area shall take place only when the reactor is shut down, the adaptor/gate valve is installed and the transfer cask is in-place.
Accommodation Approach: to be considered as a prerequisite before starting the transfer operation within the reactor module

2. The transfer of IVTM assembly into the reactor closure head from which UIS system was removed shall be done with the IVTM transfer cask(ITC).
Accommodation Approach:

The ITC shall be designed as shown in Figure 21 and the overhead crane shall be used or an alternative system shall be designed for transferring the ITC from the CV into the reactor head access area.

c) Transfer of Core Assemblies

The following operations are used to transfer core assemblies within or between fuel handling facilities.

1. Transfer operations between the reactor and RSB shall take place only when the reactor is shut down. Transfer operations within FRSSS facilities may be performed at any time.
Accommodation Approach: to be considered as a prerequisite before starting the transfer operation within the reactor module

2. The transfer of core assemblies between the reactor and the fuel cycle facility (FRSSS) shall be done with the fuel transfer cask(FTC).

Accommodation Approach:

The FTC shall be designed as shown in Figure 22 and the CT shall be used for transferring the FTC from the reactor into FRSSS.

Preparation and Transfer or Shipment of Irradiated Assemblies

The inert atmosphere fuel handling cell (FHC) shall provide for preparation of spent core assemblies for transfer or shipment as follows:

1. The FHC shall contain a handling and transporting method for moving drip dry spent core assemblies between the transfer port position and FHC temporary storage positions.
Accommodation Approach: To be designed

3.2.4 Structural Requirements

The structural design shall be based on the following:

1. A design factor of 1.5(TBD) shall be applied to the normal expected load (force) capabilities of the handling equipment to account for possible increased requirements. Safety factors shall be applied to the design factored loads.

2. Design of the RFHS shall be based on the design and normal steady-state operating conditions given in Table 6 and the design events as listed in Table 7.

3. The seismic category and method of analysis for subsystems and components are listed in Table 6. The seismic response spectrum to be used for seismic analysis shall be the building response at the location of the component support.

4. The seismic response spectrum to be used for seismic analysis of the FRSSS shall be the building response at the location of the component support.

Accommodation approach:

The structural evaluation shall be performed based on the plant duty cycles events. KALIMER duty cycles and IVTM design and its interfaces are not sufficiently established to permit development of detailed structural analysis procedures. However, the structural evaluation will follow the following general approach:

a) Duty Cycle Definition: IVTM duty cycle will be established in terms of events enveloping the IVTM transportation, plug in/out operations and refueling operations. The events will be divided into different normal and abnormal operation levels based on the IVTM performance requirements during and following the event.

b) Load Definition: IVTM seismic, transportation, thermal and refueling loads will be defined for the various duty cycle events on the basis of refueling load requirements and the reactor and transport system seismic and transportation analyses.

c) Design Criteria Development: Component stress limits will be based on the specified design criteria for the different event categories and will be supplemented by deformation limits based on the component functional and alignment requirements. The repeatability required of the IVTM positioning operation may place more stringent requirement than the structural design criteria.

d) Structural Analyses: System and component stress, deformation and alignment analyses will be performed to calculate the IVTM response to the duty cycle loads. The analyses are expected to be limited to elastic analyses in view of the IVTM positioning requirements.

e) Design Evaluation: Design margins will be calculated through comparison of the analysis results with appropriate structural and functional design limits.

Table 6 Structural Design Basis (RFHS & ITS)

Subsystem or Component	Construction Code	Seismic Category	Method of Seismic Design and Analysis	Design Conditions	Normal Steady-State Operating Conditions	
				Temperature (°C)	Temperature (°C)	Pressure (MPag)
IVTM Drive	Comm.	1	Dynamic	52	52	0.1
IVTM In-Vessel	ASME III, 1	1	Dynamic	TBD	TBD	TBD
IVTM Grapple	ASME III, 3	1	Dynamic	TBD	TBD	TBD
Plug Drives	Comm.	2	Dynamic	74	60	-
FTC Hoist	Comm.	1	Dynamic	66	49	-
FTC Bucket	ASME III, 1	1	Dynamic	816	260	-
FTC Rx Port	ASME III, 1	1	Dynamic	74-510	49-246	TBD
FTC Rx Port Plug	ASME III, 1	1	Dynamic	74-510	49-510	TBD
RSB Port	ASME III, 2	1	Dynamic	74	49	0.1

Table 7 Design Events In Reactor

Subsystem or Component	Operation	Number of Occurrences per 60 Years	Load or Force	
			Normal	Maximum Expected
IVTM assembly Hoist	Lift/Pull	50,000(TBD)*	TBD	TBD
	Lower/Push	50,000(TBD)*	TBD	TBD
	Grapple Operation	50,000(TBD)*	-	-
UIS System Hoist	Lift/Pull	TBD	TBD	TBD
	Lower/Push	TBD	TBD	TBD

* For each reactor module

3.2.5 Safety Requirements

General

The RRS shall satisfy the safety criteria as follows:

1. The safety class and corresponding codes and standards for RRS equipment and facilities shall be determined based on their safety significance. The safety classes of RRS equipment and facilities or components thereof shall be as shown in Table 8. The draft ANS 54.6 Standard, "LMFBR Safety Classification and Related Requirements," should be used as a guide in establishing these requirements.

2. The RRS design shall limit radiation exposure to 10CFR20 limits and ALARA (As Low As Reasonably Achievable) objectives during normal operation and anticipated events. The guidance in Regulatory Guides 8.8, 8.10, and 8.19 should be used to help meet ALARA objectives.

3. The RRS shall protect the health and safety of plant operators and the general public in accident situations. Table 9 contains a list of potential accidents which should be considered as a minimum.

4. The RRS equipment and facilities shall be designed to contain the gaseous radioactivity from the rupture of TBD pins.

Table 8 Reactor Refueling System Equipment Safety Classification

Equipment	Safety Class
1. IVTM In-Vessel	SC-1
2. IVTM Grapple	SC-3
3. FTC	SC-1
4. FTC-Reactor Port	SC-1
5. FTC-Reactor Port Plug	SC-1
6. FTC-FHC Port	SC-2
7. FHC Handling Equipment	SC-2
8. UTC	TBD
9. ITC	TBD

Table 9 Unusual Events To Be Considered In Design And Analysis Of The RRS

Event
- Random fuel rod leakage in equipment or facility
- Single operator error
- Failure of any single active component
- Malfunction of a RFHS grapple requiring cleaning or repair
- Loss of all off-site power
- OBE
- SSE
- Inability to release a grappled component
- Failure of a grapple, requiring replacement
- Dropping of a component by hoisting equipment(including UTC and ITC)
- Cover gas purification system failure
- Insertion of incorrect assembly into empty position or an assembly into an already occupied core position.
- Dropping of a core assembly into storage from a height exceeding 7.6 cm(TBD)
- Dropping of UIS system
- Dropping of IVTM assembly
- Immobilization of heat-producing core assembly during normal operations
- Storage of up to 1% defected fuel

5. The passive barrier between spent fuel handling operations and the environment shall be designed to reliably maintain its leak-tight integrity during normal operations and in accident situations. Radiation detectors shall be provided to detect significant radioactivity leakage through this barrier. Multiple seals with leak monitoring in the seal space shall be used for critical sealing functions.

6. The fuel handling system shall be designed to maintain fuel in a safe condition during and after all plant design basis accidents and following postulated evacuation of refueling personnel.

Criticality

The RRS design shall preclude attaining $K_{eff} \geq 0.95$ during both normal and malfunction conditions. Materials having good neutron moderating properties shall not be used in RRS equipment or facilities. RRS facilities in which an array of new or spent fuel assemblies can be stored shall be provided with criticality monitors. These monitors shall alarm if a condition of criticality is being approached.

Fuel Handling Equipment

Fuel handling equipment shall be designed to preclude dropping fuel elements. The requirements in the NRC Standard Review Plan, Section 9.1.4, "Fuel Handling System," and NRC Branch Technical Position APCSBP-1, "Overhead Crane Handling Systems for Nuclear Power Plants," shall be used where applicable.

Hard stops shall be provided at the end of the motions of transfer machines to prevent collision of the transfer machine with other objects.

UIS System Handling Equipment

UIS system handling equipment shall be designed to preclude dropping UIS system. The requirements in the NRC Standard Review Plan, Section 9.1.4, "Fuel Handling

System," and NRC Branch Technical Position APCSBP-1, "Overhead Crane Handling Systems for Nuclear Power Plants," shall be used where applicable.

Hard stops shall be provided at the end of the motions of transfer machines to prevent collision of the transfer machine with other objects.

IVTM Assembly Handling Equipment

IVTM assembly handling equipment shall be designed to preclude dropping IVTM assembly. The requirements in the NRC Standard Review Plan, Section 9.1.4, "Fuel Handling System," and NRC Branch Technical Position APCSBP-1, "Overhead Crane Handling Systems for Nuclear Power Plants," shall be used where applicable.

Hard stops shall be provided at the end of the motions of transfer machines to prevent collision of the transfer machine with other objects.

Accommodation approach:

The safety evaluations shall be performed to prove the fulfillment of the safety criteria in case of design bases accidents of the RRS. Also some functional equipments shall be designed to preclude the possibilities of the accidents functionally expected.

3.2.6 Instrumentation and Control Requirements

Overall Refueling Control System

All functions performed by the RRS shall be controlled and monitored by an integrated overall refueling control system designed to support both normal and infrequent fuel handling operations. There shall be operations control over vital components to prevent unauthorized manipulation of said components by insiders. Access and operations control together address collusion threats.

The normal mode of control shall be by operator supervision of computer controlled sequences. The computer shall also be used for directing the operator sequences to be used, and shall contain pertinent data useful to the operator, including the refueling plan.

Local control consoles shall be provided for the purpose of checkout, maintenance, or emergency operation of the RFHS components. These consoles shall be either located on the equipment or nearby where the operation can be directly viewed or viewed with TV.

Implementation of Refueling Plan

The refueling control system shall be capable of accepting and implementing a refueling plan and of accepting changes to the plan during the course of normal refueling operations.

The refueling plan shall contain a list of core assemblies, or groups of assemblies to be changed, and shall include: (1) the type, (2) the serial number, (3) the present location, (4) location after refueling is complete, (5) orientation of core assembly in core, (6) sequence in which core assemblies are to be moved, and (7) any special conditions.

A detailed refueling procedure for use by the operating personnel shall be prepared. It shall consist of all moves performed under automatic or manual control during the refueling operation and shall include specific reactor core positions and storage positions. The order of component replacement shall be chosen to satisfy the requirement of the refueling plan and to expedite refueling.

Inventory Control

This system should (1) perform nuclear materials measurements, (2) perform data analysis to account for nuclear material, (3) maintain records and provide reports, (4) assign and exercise responsibility for nuclear material, and (5) monitor internal movements, location, and utilization of nuclear materials as listed below.

1. Inventory control is defined as maintaining a record of the current location of all core assemblies on-site. A record of special core assemblies shall also be maintained.

2. A record of all core assemblies shall be maintained in an inventory control system having mutually independent entries and a process for noting and resolving discrepancies.

3. Core assemblies entering the plant shall be individually identified visually and recorded in the inventory manually and automatically as part of new fuel receiving inspection.

4. Core assemblies leaving the plant shall be physically identified as part of spent core assembly inspection. Identification data shall be entered into the inventory control system both manually and automatically.

5. Essential data to determine inventory changes in the nuclear materials shall be recorded automatically, with manual backup capability.

6. The inventory data shall be converted into an acceptable format and transmitted by the plant data handling and transmission function of the plant control system. Inventory data shall be stored.

Load Control

Push and pull loads on components handled shall be measured and controlled so that loads in excess of that normally required may be applied only under strictly controlled conditions. If loads above normal are applied, the peak load shall be recorded. Loads above normal shall actuate an interlock blocking planned operation until a supervisory release is input.

Controls and Interlocks

Controls and interlocks shall be provided to prevent accidents and to minimize plant unavailability. Controls and interlocks shall be fail-safe. Mechanical interlocks shall be used wherever possible.

a) Interlocks

Interlocks used to prevent events, or to mitigate the consequences of such events, which could result in: (1) failures endangering plant operating personnel, (2) significant plant or refueling unavailability, or (3) significant equipment or facility damage, shall be hard-wired.

All other interlocks may be accomplished by firmware or software.

Ideally, components involved in interlocks shall be completely independent of components performing control, monitoring, or other control system functions. Where this is not cost effective, these interlocks may be combined with control circuits, provided that protection will not be inhibited as a result of operation or failure of the control equipment.

Failure of Automated Equipment

a) Computer Failure

Failure of any automatic control equipment (e.g., the computer) shall not prevent refueling from proceeding. All control functions using computers, including sequence control and inventory control, shall also be provided with manual control. Under Manual control, actions shall be initiated manually, but may be controlled either manually or by the use of automatic controllers that are not dependent upon computers.

b) Backup Manual Control

Manual controls shall be provided as backup for equipment and facilities which are automatically controlled. The manual control point shall be located in view of the equipment or facilities being controlled when it is advantageous, and directly viewable; otherwise, the manual backup controls shall be in the refueling control console (RCC).

c) Computer Memory

The computer control system memory shall not be lost on loss of power and shall retain its data unaltered. Resumption of power following an outage shall permit resumption of computer operation.

Power Requirements

a) Available Voltages

Individual electrical equipment items of the refueling control system shall be designed to operate at one of the voltages supplied by the building electrical power system, If electrical equipment require voltages or regulation not supplied by this system, it shall be necessary for the control system to provide any additional equipment to obtain such voltages or regulation.

b) Emergency Power

RRS loads on the non-Class 1E Uninterruptable AC Power System (UPS) shall be strictly limited to those needed for transmitting alarms to the plant security system (per 10CFR73) and to those needed to prevent the loss of normal power due to faults endangering plant operating personnel or significant equipment or facility damage.

Accommodation Approach : The approaches for the instrumentation and control system shall be performed in conceptual design stage.

3.3 System Performance Characteristics

The RRS performs the normal operating modes, infrequent modes, and off-normal modes.

During normal refueling of the reactor, the TS and FRSSS must support the RFHS to achieve a refueling rate so that at normal working efficiencies the reactor refueling can be completed within the allotted 30 days.

For the KALIMER plant of three modules, refueling of modules will be in progress the good portion of the TBD period and therefore the FRSSS will be continuously in the process of receiving and preparing new fuel for the next refueling outage along with transferring spent core elements as they are discharged from each reactor. During the time between module refueling, the fuel handling equipment will be maintained to assure proper operation.

Normal Operating Mode

The RFHS will normally operate according to a planned schedule of core assemblies to be replaced. The actual order of assemblies to be replaced will be determined by reactor operations. The spent control assemblies are exchanged first to increase core negative reactivity. The lower heat-producing spent fuel assemblies following a TBD month storage are exchanged, then the radial shields, and last, the higher heat-producing spent blanket assemblies are exchanged in reverse order of decay power. As a consequence of this refueling sequence, time is permitted for the reduction of decay heat of the blankets to permit drip dry removal without in-vessel storage time. The normal maximum fuel assembly from a previous refueling outage will be producing about TBD kW of decay energy. There is no fuel assembly decay heat problem in the reactor vessel or in the transferring the fuel assemblies TBD at a time in the FTC in the drip dry condition. Radial Blanket assemblies will be transferred TBD at a time in the FTC in the drip dry condition while internal blankets will be transferred one at a time.

IVTM Infrequent and Unusual Operation

A) Dimensional gauging of the reactor internals: The periodical plan for the KALIMER design is to use the IVTM for dimensional gauging of the IVTM assembly plugged and the reactor internals. Grappling points at certain reactor internals locations(for example, top of the core former ring, fuel storage racks and others) would be established. The IVTM would grapple these points and the X, Y and Z coordinates of the IVTM grappling head recorded. This would be repeated periodically and evaluated for shifting or distortion of the IVTM assembly and reactor internal structures.

B) Inlet module replacement: The IVTM is designed to replace the core inlet modules. This requires removal of the core assemblies and use of a special module grappling tool. Replacement of the inlet model is not planned. However, this capability is imposed on the IVTM *in the event it is needed.*

C) Recovery from IVTM malfunction: The IVTM is designed to recover from single failure malfunctions. In the event one of the IVTM functions fails, the remaining operable functions are used to release a grappled fuel assembly, the IVTM pantograph is collapsed and the IVTM removed from the reactor. Examples of this are shown in the attached figures.

3.4 Impact on The Plant Availability

Plant availability is defined as the probability that an item or system will be operational on demand, in other words, the ratio of the time available for power operation over the total time. Equivalent availability is defined as the the percent of electricity produced by a plant compared with the amount it will produce if operated at full power during the entire period. The capacity factor would equal the equivalent availability if the maximum available power output of the plant were always in demand.

The plant availability estimate for KALIMER is performed by using the same procedure as PRISM. Also the plant parameters such as failure rate and averaging repair time for each system component, are assumed to be same as those used in PRISM.

The plant availability estimation incorporates several notable design features, a result of trade studies and other evaluations. The design features, which potentially impact on plant reliability and availability, are listed below:

- o An 24 month refueling cycle was adopted.
- o Fuel transfer cask capacity was six core assemblies
- o The refueling operations were assumed producing a refueling outage 30 days or 32 days
- o As a steam generator reference, a single wall helical coil steam generator was assumed similar to PRISM case.
- o The seismically isolated structure includes both the EM pump synchronous coastdown machines and the reactor protection system(RPS) equipment

- o The major & minor overhaul outage of the entire turbine-generator, which will be performed once every two years(major: every 6 years, minor: every 2 years), was set at 17 days by using a spare rotor allowing off-line rebuilding.
- o The in-service-inspection(ISI) interval(57 days) for steam generator tubes was set at 10 years per SG unit. This interval will provide a SG inspection interval of 10 years per power block

The availability block diagram for KALIMER is depicted in Figure 24. The combination of one NSSS(Nuclear Steam Supply System) and one TGI(Turbine Generator Island) is called a power block(PB). Three power blocks constitute the KALIMER plant. Each PB is an independent power plant capable of supplying electrical power to the utility grid. The power output of a PB is 333MWe. The power blocks operate as three independent power generating units since the steam lines fed by the NSSS units are not interconnected across the PB envelopes. The common thread that connects the PB to form a single plant, aside from location on the same site, is that all reactors are controlled from a common control room. Therefore the plant incorporates three NSSS units(three reactor modules) and three TGIs(three TG sets). The combined power output at the plant level is 1000 MWe.

To assess the plant availability, an operating scenario should be determined and modeled. The Markovian process is introduced in this report for modeling. The detailed description of the Markov modeling technique and procedure is described in the Chapter 17 of 'LMR technology'.

Figure 25 shows the operating scenario for the one PB of the KALIMER plant. Two different refueling outage periods(30 and 32 days) are assumed because the additional process time due to UIS plug out and IVTM plug in is variable. The 30 days are determined by adding 5 days to the PRISM refueling outage period(25 days) and The 32 days one week. The corresponding Markov transition diagram is presented in Figure 26. This diagram show the relationship between the competing failure and repair processes.

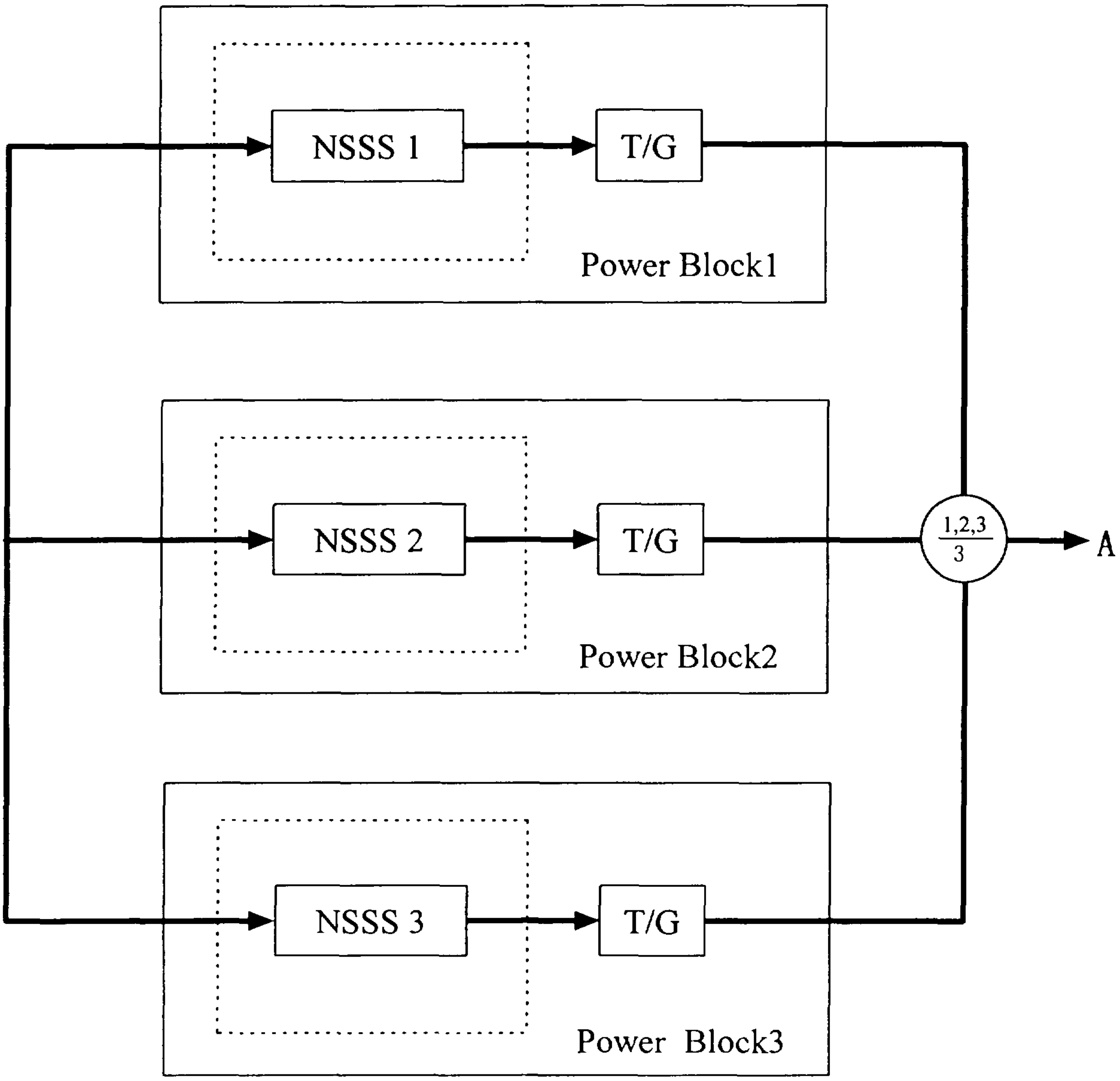


Figure 24 KALIMER Power Block Diagram

The failure and repair rates data for each NSSS component and TGI are assumed to be same as the PRISM plant. The plant equivalent availability factor(EAF) has been estimated by using MARK code which is a computer code for simulating Markov processes. The scope of this estimation is limited to only those outages which are a consequence of the following:

- o Unplanned outages suffered as a consequence of equipment failures.
- o Planned outages associated with performance of scheduled preventive maintenance.
- o Planned outages associated with reactor refueling operations.

The reactor outages which result from unavailability of new fuel are not reflected in the results presented in this report. The failure and repair events which characterize the PB from an availability standpoint are assumed to be statistically independent, then the plant equivalent availability factor for the plant and PB are equal.

Table 10 and 11 summarizes the results of a code calculation of the Markov models for two different refueling outage intervals(30 and 32 days). It shows that the availabilities of KALIMER plant are calculated to be 91.99% and 91.78%, respectively. Those are a little bit lower than PRISM case(92.5%). Accordingly it is turned out that the KALIMER plant availability becomes slightly lower than the PRISM one(92.5%) and thus the impact of the KALIMER refueling system on the plant availability is negligible.

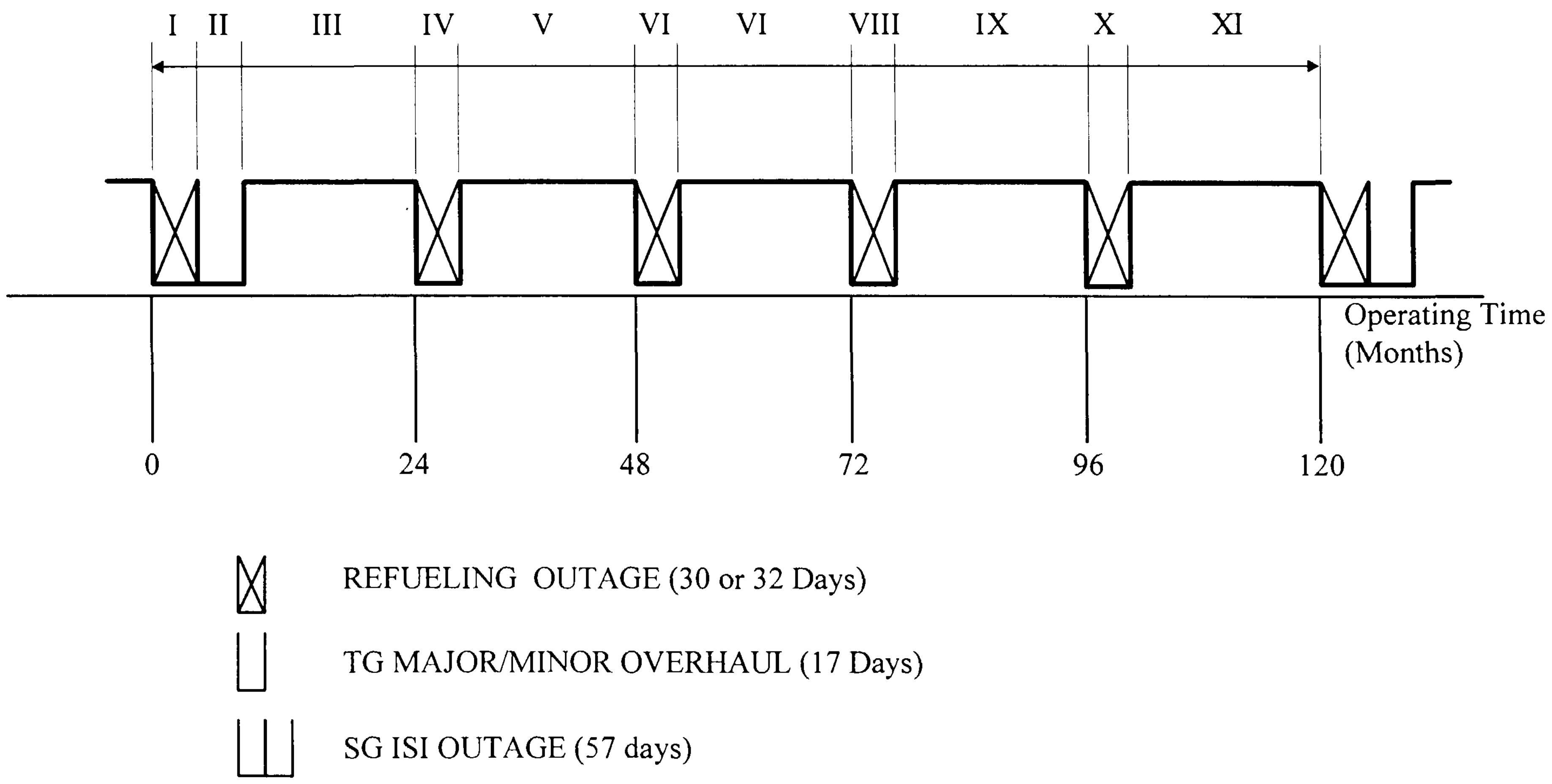
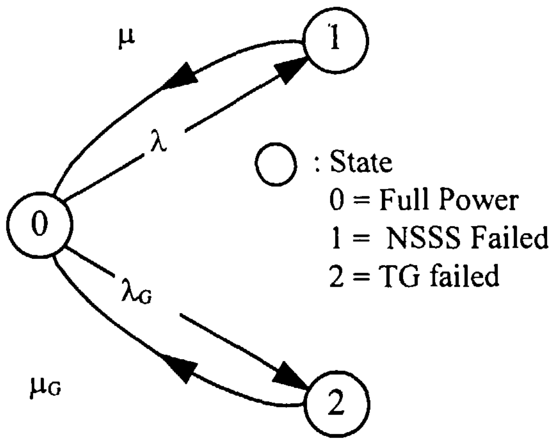


Figure 25 Operating Scenario for One Power Block



$$\mathrm{EAF} \cong \left(\frac{\mathrm{MTTR}}{\mathrm{MTTR}+\mathrm{MTTF}}\right) \times 100$$

$$\mathrm{MTTF} = \frac{1}{\lambda_T}$$

$$\lambda_{\mathrm{T}} = \sum \lambda_{\mathrm{i}} = \lambda + \lambda_G$$

$$\mathrm{MTTR} = \frac{\sum \lambda_{\mathrm{i}} \mathrm{T}_{\mathrm{i}}}{\sum \lambda_{\mathrm{i}}} = \frac{1}{\mu_T} = \frac{\frac{\lambda}{\mu} + \frac{\lambda_G}{\mu_G}}{\lambda + \lambda_G}$$

where

EAF: Equivalent Availability Factor

MTTF: Meantime to failure

MTTR: Meantime to repair

λ_{i}: Failure rate

μ_{i}: Repair rate

Figure 26 Markov model for KALIMER plant

Table 10 KALIMER Plant Availability(case I)

Availability Analysis of KALIMER Plant (**30** days for refueling outage)			
Sequence	Total Duration (hours)	100% Power Operation (hours)	0% Power Operation (hours)
I+II	1,368	0	1,368
III	16,152	15,616	536
IV	720	0	720
V	16,800	16,242	558
VI	720	0	720
VII	16,800	16,242	558
VIII	720	0	720
IX	16,800	16,242	558
X	720	0	720
XI	16,800	16,242	558
Total	87,600	80,584	168,184
Availability	91.99%		

Table 11 KALIMER Plant Availability(case II)

Availability Analysis of KALIMER Plant (**32** days for refueling outage)			
Sequence	Total Duration (hours)	100% Power Operation (hours)	0% Power Operation (hours)
I+II	1,368	0	1,368
III	16,152	15,616	536
IX	768	0	768
V	16,752	16,196	556
VI	768	0	768
VII	16,752	16,196	556
VIII	768	0	768
IX	16,752	16,196	556
X	768	0	768
XI	16,752	16,196	556
Total	87,600	80,400	7,200
Availability	91.78%		

4. Conclusion

The pre-conceptual design of the KALIMER upper internal structure(UIS) and reactor refueling system has been described. For the UIS system, the functional, structural and material requirements have been determined and the accommodation approaches to meet these functional requirements described. For the refueling system, the functional, structural, process and I&C (Instrument and Control) requirements have been established and the accommodation approaches for the functional and process requirements described. The impact on plant availability due to extension of the refueling outage has also been investigated.

Up to now, the followings have been concluded:

1) The accommodation approaches for UIS system show that the design concept of the system will satisfy the functional requirements with a few design issues to be resolved, such as UIS plug in/out handling system and cask design.
2) The functional and process requirements of the refueling system are achievable with the design of the IVTM cask and related transfer system.
3) The extended refueling outage has little effect(within 1 %) on the plant availability if extra refueling time do not exceed 1 week.

Reference

1) Preliminary Safety Information Document, General Electric Co., 1986

서 지 정 보 양 식

수행기관보고서번호	위탁기관보고서번호	표준보고서번호	INIS 주제코드
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연 구 자 및 부 서 명	박진호 (응용연구그룹 액체금속로냉각재설계기술개발분야)

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연구위탁기관		계약 번호			

초록 (15-20줄내외)

KALIMER 노상부 구조물(Upper Internal Structure) 및 핵연료 교체 시스템에 대한 설계 연구를 수행하였다. 두 시스템의 뚜렷한 특징으로서는 착탈식 노 상부 구조물의 개념 도입 및 이로 인한 핵연료 교체 기간의 증가이다. 노 상부 구조물에 대해서는 기능, 구조, 재료 측면에서의 설계 요구 사항 정립 및 기능적 요건의 수용을 위한 개념적인 접근 방법을 도출하였다. 핵연료 교체 시스템의 경우에는 기능, 구조, 공정 및 I&C(Instrument and Controls) 측면에서의 설계 요구 사항을 마련하였으며 기능 및 공정 요구 사항을 수용하기 위한 방안을 개념적으로 정리하였다. 아울러 핵연료 교체 기간 증가로 인한 발전소 활용도(availability)에 대한 영향을 GE사의 Markov code를 이용하여 평가하였다.

착탈식 노 상부 구조물의 경우에는 착탈을 위한 부가적인 기구 및 캐스크 설계 문제가 해결되면 설계 요구 사항을 대부분 만족시킬 수 있을 것으로 판단되며, 핵연료 교체 시스템의 경우에는 기능 및 공정상의 설계 요구 사항은 IVTM 캐스크 및 이동 관련 계통의 추가를 고려하면 설계상으로 특별한 문제점 없이 만족될 수 있을 것으로 사료된다. 그리고 노 상부 구조물의 착탈로 인한 핵연료 교체 기간의 증가가 1 주일을 초과하지 않을 경우, 발전소의 활용도에 대한 영향은 무시할 수 있는 것(1% 이내)으로 나타났다.

주제명키워드 (10단어내외)	노 상부구조물, 착탈식 플러그, 핵연료 교환기, 핵연료교체주기, 발전소 활용도

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Abstract (15-20 Lines)

The design study for the KALIMER upper internal structure(UIS) and reactor refueling system has been described. Two distinct features are plug-in UIS and extended refueling outage. For the UIS system, the functional, structural and material requirements have been determined and the accommodation approaches to meet these functional requirements described. For the refueling system, the functional, structural, process and I&C (Instrument and Control) requirements have been established and the accommodation approaches for the functional and process requirements described. The impact on plant availability due to extension of the refueling outage has also been investigated.

The accommodation approaches for UIS system show that the design concept of the system will satisfy the functional requirements with a few design issues to be resolved, such as UIS plug in/out handling system and cask design. It is also shown that the functional and process requirements of the refueling system are achievable with the design of the IVTM cask and related transfer system and the extended refueling outage has little effect(within 1 %) on the plant availability if extra refueling time do not exceed 1 week.

Subject Keywords (About 10 words)	Upper Internal Structure, Plug out/in type, In-vessel Fuel Transfer Machine, Reactor Refueling System, Plant Availability