ARC-100 Facility

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WHITE PAPER ON SPENT FUEL STORAGE INSIDE THE REACTOR VESSEL

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1. INTRODUCTION

This white paper describes the approach to be utilized in the ARC 100 facility for storage of spent fuel. The ARC 100 normal refueling cycle is 20 years. During that time, reactor shutdowns, will periodically occur, with the reason for the shutdown determining whether it is a hot-shutdown (for example analysis of an unexpected reactivity signal in the control room) or cold shutdown (examples include replacement of the control elements neutron absorbers, in service inspection, and measurement of pull-out forces in assemblies). The driver fuel assemblies, however, will normally remain in the core for 20 years, and be irradiated for 18 equivalent full power years, assuming a capacity factor of 90%. While not directly relevant to the storage , Appendix A provides the estimated capacity factor of the ARC 100 facility, as it establishes the total irradiation to which the fuel is subjected.

At the end of the refueling cycle, fresh fuel replaces the spent driver fuel, which must be safely stored until ultimately being disposed. The choice of the interim storage of the spent driver fuel (and potentially of a limited number of driver fuel assemblies, which may need to be replaced because of identified cladding failures) is determined from considerations of fuel overall performance, safety in handling the fuel, security and safeguarding of the fuel, operational ease and flexibility, duration of refueling outages, and ultimate decommissioning of the facility.

1.1 Description of advantages and disadvantages of storing spent fuel inside the Reactor Vessel

The ARC 100 reactor is designed to be refueled every 20 years. Therefore, during the life of the facility (designed for 60 years), their normal refueling will occur at year 20 , and 40 with a final defueling at year 60. The fuel handling and storage systems are designed to be safe whether the fuel is stored in vessel or ex-vessel or a combination of the two. However, from the safety, safeguards and security standpoint, as well as operational, decommissioning and economic standpoints there are relative advantages and disadvantages to the handling and storing of the spent fuel in vessel and ex-vessel.

1.1.1 Advantages

A. Safety

Storing spent fuel in-vessel results in handling the fuel within the reactor vessel when removing each furl assembly from the core in the two normal refueling operations by transferring it to the in-vessel storage location, where the spent fuel normally remains until the next refueling outage , at which time the fuel will have decayed for 20 years.

At 1st refueling (Year 20) in case of In-vessel storage, the In-Vessel Transfer Machine (IVTM) removes "dummy" assemblies one by one from the in-vessel storage locations, where they were stored during the initial fueling operations [Note 1], and moves them one by one to the in vessel location under the transfer port (designated as the Transfer Location-TL).

The IVTM then retrieves the spent fuel assemblies from the core, one by one, and transfers them to the in-vessel storage locations vacated by the removal of the "dummy" assemblies.

[Note 1] *An option is to store the dummy assemblies out of the reactor vessel prior to initiating operations. This would have the advantage of storing assemblies that have not been activated, and of shortening the first refueling. The* *disadvantage is not utilizing the dummy assemblies as part of the core barrel shielding. When stored in vessel, the dummy assemblies reduce the amount of additional shielding needed to limit activation of the secondary sodium, NaK in the DRACS and air in the RVACS. A final decision has not been made.*

The Fuel Unloading Machine (FUM) retrieves those "dummy" assemblies out of the vessel and into a transportation cask for movement to a storage location. The dummy assemblies are activated, thereby requiring a shielded area for storing, but require no active cooling. The FUM then picks up a fresh fuel assembly from the storage rack within the reactor building and lowers it to the Transfer Location . The IVTM retrieves it from the TL, places it in the core, and moves to retrieve another dummy assembly. The operation of the IVTM and FUM continues until all of the dummy assemblies have been retrieved, all of the spent fuel of the first core has been stored in the vessel storage locations, and the fresh fuel has been loaded in the core. No spent fuel will normally leave the reactor vessel until the second refueling. For the 2nd refueling (year 40) the same operations take place, but instead of "dummy" assemblies, the IVTM and FUM handle spent fuel assemblies, and the entire fuel of the first core is transferred out of the reactor vessel and transported to the on-site dry cask storage facility. Appendix B provides a time and motion study describing in more detail the operations that take place when the spent fuel is stored in-vessel. In the 2nd refueling, the FUM operation takes place at a time when the very low decay heat of the fuel (20-year decay) minimizes the likelihood of an incident resulting in fuel damage and release of radioactivity in case the fuel assembly becomes stuck during the transfer out of the vessel or the transfer to the transportation cask that moves it to the onsite dry storage facility. The fuel has decayed long enough to allow immediate placement in the onsite dry cask storage facility. In fact, disposal of a spent core in the onsite dry cask storage facility is possible as early as 4 years after shutdown .

For the final de-fueling, there is an option of moving the last core to the in-vessel storage location, in which case the operations for the spent fuel are identical as those for core no 2, but it is much more likely that it the last core will be allowed to cool in the core, and only the spent fuel stored in the in vessel storage location is moved to the on-site dry cask storage facility.

Unless a lengthy period elapses between shutdown and spent fuel handling, storing ex-vessel requires removing the spent assembly from the core, and transporting it to the ex-vessel storage facility at times when the decay heat is still quite significant, increasing the consequences of an accident resulting in radioactivity release. While the FUM will be designed to accommodate spent fuel with relatively high decay heat, there is a practical limit (~ 5KW) to the active heat removal (see note 2), and meeting that limit requires the fuel to have decayed at least 15 days (see note 3). If stored in a separate tank, the fuel is handled once to move it to the external tank and then again from the external tank to the transportation cask, essentially resulting in the fuel being handled more times, which qualitatively increase the possibility of an incident.

Note 2 The FUM is designed to provide pre-heating of fresh fuel assemblies to about 230°C to prevent thermal shock when the fresh fuel assembly is immersed in the sodium pool, and to provide cooling of the spent assembly to limit heating the fuel pins to a temperature below 550°C. The cooling system employs forced very pure Argon, which is also utilized to blow off residual sodium adhering to the fuel assembly. Providing an argon flow through

the ARC 100 assemblies capable of removing heat significantly in excess of 5 KW becomes increasingly impractical as pressure drop become too high, given the dimensions of the fuel assembly (length in excess of 5 meters) and space between the duct and the pins [small equivalent diameters(De~0.01cm)]. Of course, it is possible to design for higher heat loads, but active systems can fail, and a system relying on natural circulation is far more preferable.

Note 3 The 15 days would not apply to assemblies that have the peak decay heat, but to assemblies having a lower heat.

B. Security and Safeguards

In vessel storage offers the advantage of having the fuel in the seismically isolated reactor building, with infrequent access of personnel compared to the access in another building. Storing in an external tank requires security and accountability at two locations : reactor building and location of the external storage tank.

C. Operations

Storing in vessel requires the IVTM, the FUM , a wash station (to remove sodium from the fuel) and a transportation cask that interfaces with the on-site dry storage facility. Only the transportation cask moves between the reactor building and the on-site dry storage facility. The FUM is only used within the reactor building.

Storing ex-vessel requires an additional transportation cask designed for single assembly transfers, plus the external tank and its heat removal equipment , and either the ability of the FUM to move from the reactor building to the building housing the external storage tank, or a separate fuel handling component to move the assembly in and out of the tank.

In addition, in-vessel storage results in shorter refueling intervals which can contribute to a higher capacity factor. The difference is addressed in Appendix B: ”Time and motion study comparing ex-vessel fuel storage and in vessel fuel storage” In vessel storage results in refueling outages effectively at least 15 days shorter than storing ex-vessel.

Storing in-vessel eliminates the need for an ex-vessel storage tank with its heat removal and monitoring equipment. Consequently, operations and maintenance activities that would be associated with the external storage are eliminated.

Although not directly relevant from the safety, security/safeguards and operations standpoint, there is also an economic advantage to storing in vessel, because that option does not necessitate the external storage tank, its heat removal system, the additional quantity of sodium and the facility house them.

D. Decommissioning

To decommission of one vs. two vessels is an advantage, as is having to process significantly lower quantities of liquid sodium and fewer components that are sodium and possibly radioactively contaminated.

Given that safety and operational standpoints offer advantages which outweigh the possible disadvantages summarized in 1.1.2 , it is also germane to know that the costs of storing in vessel will be less than the cost for storing ex-vessel, with the difference being driven by two major factors: the absence of an external storage tank with its preferably passive (or active) cooling systems, and the facility that would house the external tank and other fuel handling components.

1.1.2 Disadvantages

A. Operational

The most obvious disadvantage in storing the spent fuel within the reactor vessel is the inability of moving the spent fuel from the storage tank to the on-site dry storage casks (or the repository) without shutting down the reactor. Storing the spent fuel in an ex-vessel storage tank, decouples the reactor operation from the stored fuel operations, with the latter proceeding whenever practical, thereby making more storage available within the tank, as spent fuel is transferred to the outside dry storage facility. This provides more flexibility for storing defective fuel in the external storage tank, than might be available in the limited spare locations within the in-vessel storage racks. The in-vessel storage locations are limited to the driver fuel assemblies (99), control rods (6 controls and possibly 3 safety) and a few potentially defective assemblies. Once those locations are full, moving additional assemblies would require shutting down the reactor and removing however many assemblies are required to create the needed vacant position.

The likelihood of significant fuel failures is small, and the emptying numerous in-vessel (half or all) stored assemblies can be done coincidently with minor (earliest planned in year 4) or major outages (earliest planned year 8) for equipment refurbishment. Since the decay heat of the assemblies has already reduced more than sufficiently for storage in dry cask (assemblies at year 4 have a decay heat of about 0.7 kW and at 8 years 0.5kW), the spent fuel can be moved to the dry cask facility, creating additional vacant space for defective assemblies or possibly assemblies which are judged to have deformations that left uncorrected over time could lead to difficulties in extraction from the core and for which storage is determined to be preferable to rotation .

In fact, movement to the dry storage facility could begin in approximately 7 months after shutdown, when the assembly decay heat falls below 2.0 kW, but transfer by the FUM would require active cooling systems with a possibility of an incident, if the active system were to fail or if fuel assembly where to get stuck in between the reactor vessel and the interior of the FUM with limited cooling flow. Therefore, the risk that an unplanned shutdown of long duration would be required is small, and therefore this is not a large disadvantage. Figure 1 shows the decay heat of an ARC-100 assembly as a function of time after shutdown.

B. Safety

From the standpoint of accidents, a dropped fuel assembly within the vessel is much more difficult to recover than a fuel assembly dropped in a separate storage vessel. Such assembly drop can occur only during fuel handling and motion within the vessels, and storing in vessel involves twice the number of fuel movements within the reactor vessel than storing it in an ex-vessel tank (for storage in vessel the fuel is first moved to the storage location, and later it is moved from the storage location to

the location of the FUM port, whereas for storage in an ex-vessel tank, the fuel is moved once from the core directly to the FUM port location). Therefore, the possibility of a fuel drop which would shut down the facility for an undeterminable time, is greater for in-vessel storage than for ex-vessel storage, and this presents the greatest disadvantage . Offsetting this disadvantage, however, is the overall handling of the fuel in and out of the vessels, which is greater for the ex-vessel storage as stated in Section 1.1.1

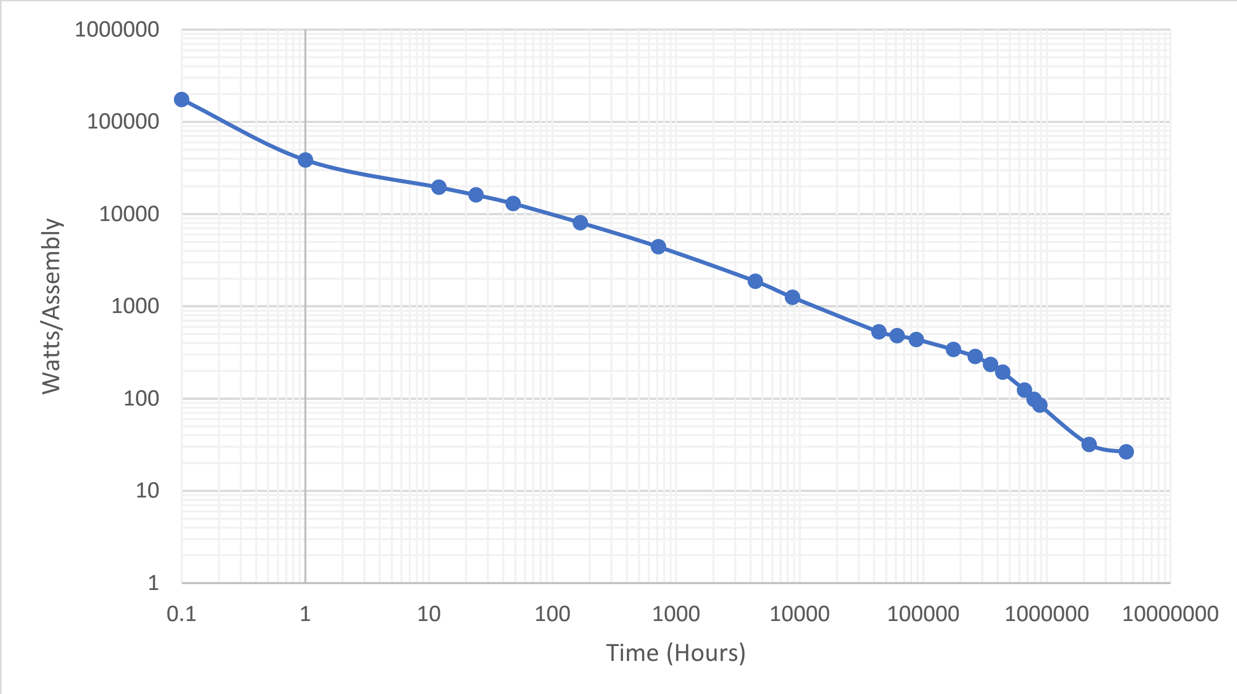

Figure 1 Assembly Decay Heat after Shutdown

2.0 FUEL PERFORMANCE

2.1 Fuel storage configuration in the vessel

[[Figure A.2 in Annex 1]][EXPC] shows both a plan view of the storage locations configuration in the reactor vessel and an elevation view taken at the location of the FUM port (the thimble at the location of the FUM Port remains empty so an assembly can be transferred from its storage position to the location from which the FUM can remove it from the reactor vessel and transfer it to the transportation cask).

The thimbles are extensions of the hot pool into the cold pool, connected at the top to the bottom of the redan, and supported at the bottom by an extension of the core support plate. A small hole at the bottom of each thimble allows cold pool sodium to penetrate the thimble and utilizing the decay heat of the fuel assembly generates sufficient flow to maintain proper cooling of the assembly.

The storage of the spent fuel in-vessel adds decay heat to the heat generated by operation of the replacement core, and this additional heat must be considered in the kinetic and thermal hydraulic performance of the replacement core and reflected in the deterministic safety analyses of the combined replacement core and stored core. The results of bounding deterministic safety analyses are reported in Section 2.4

The configuration of the stored spent core must not create the possibility of criticality, including consideration of the effects of neutron emission from the spent fuel. The analysis of the configuration is provided in Section 2.2

2.2 Criticality and Shielding Considerations–

Per DOE Order 420.1C and ANSI/ANS 54.2-1985, the storage configuration has been reviewed for critical consideration and found to be acceptable. Annex 1 presents an approximate calculation performed with one-group cross sections, neglecting, for conservatism) the fact that the majority of the storage position s are located within the shielding of the core barrel, and its result confirm that the configuration should be subcritical, with a k_{eff} less than 0.7 (0.63). However, it is recommended that a more detailed analysis be performed with multigroup cross section and an appropriate code.

2.1.2 Effect of neutron emission from spent fuel

Effect of neutron emission from spent fuel should also be considered in determining the shielding required to prevent unacceptable activation of the secondary sodium, the NaK of the DRACS and the air in the RVACS. Those effects have not yet been determined but are expected to be negligible when compared to those caused by the active core.

2.3 Spent Fuel Cooling

In-vessel stored spent fuel is immersed in an extension of the hot pool. Consequently, the spent fuel will experience higher temperature than if it were stored in an external sodium vessel, where the temperature would be close to 300 °C lower. Additionally, the decay heat generated by the spent fuel will contribute to the hot pool heat load. Therefore, two aspects are considered:

1) Is the temperature of the decaying fuel assembly at its peak decay heat level when stored acceptable? Acceptability is judged in terms of margin to fuel melting , thimble sodium boiling and cladding failure. Analysis of the maximum temperatures in the fuel and cladding under the flow established by the natural circulation within the thimbles indicate ample margins to fuel melting and sodium boiling. However, margins to cladding failure can only be determined by considering damage mechanism which may have already affected the cladding during the time the assembly was irradiated in the active core. These potentially life limiting phenomena are addressed in section 2.4.

2) Is the ARC-100 emergency heat removal system (the combination of DRACS and RVACS) capable of removing the combined decay heat of the active and stored cores, without significantly affecting the margins of safety in the active core? The consequence of having additional decay heat on the safety of the reactor is addressed in section 2.5.

2.4 Effects on Fuel Life Limiting Phenomena

The stored spent fuel will have been subjected to possible damaging effect that may have weakened the cladding. These effects are permanent strains experienced during irradiation for 20 years, loss of cladding thickness from corrosion, formation of fission product migration layers at the fuel cladding interface, and formation of eutectics. In addition, there may have been deformations caused by swelling under the neutron fluence accumulated in 20 years. The spent fuel will experience some of these phenomena during the additional 20 years of storage, and because the in-vessel storage will be at a higher temperature than if the spent fuel were stored in an external vessel, it may be expected some additional damage could occur, whereas more damage in an external storage tank is unlikely.

Analyses of the effects on the spent fuel assemblies stored in vessel for a period of 20 years indicate that additional weakening of the cladding by either eutectic formation or fuel cladding chemical interaction caused by fission product migration are small and acceptable. The temperatures of the fuel and cladding of the stored assemblies during normal operation are initially the same as those experienced in the core during hot standby (operation at decay heat removal only) and decrease over time after the initial placement in storage. These temperatures (experienced at the fuel cladding interface) are insufficient to cause eutectic formation, and the additional growth of brittle layers (over and above the thickness experienced during irradiation in the core) has been conservatively estimated using the equation below [1] to be 25.5 µm (assuming the interface temperature remains a constant 560 °C)

[[ ]]P (Eq.1)

where δ is the layer thickness in meters, t is time in seconds, and T is temperature in °K.

The same equation had been used to predict the layer thickness during the 20-year period of irradiation at a constant temperature of 613 °C (the highest 3σ temperature of the fuel cladding interface at steady state, and for those 20 years assuming 100% capacity (instead of the expected 90% capacity – see Appendix A) the FCCI layer is predicted to be 102 µm. To verify the appropriateness of the simplified equation, results obtained for a temperature of 542°C (17 µm) have been compared to a detailed BISON analysis, which at a cladding temperature of 542°C predicted a growth of 25 µm.[2] The combination of surface scratches (12.5 µm), FCCI strengthless layer thickness (127.5) and an assumed 10 µm loss due to corrosion, reduce the load bearing thickness of the cladding to 0.35 mm, with a resulting maximum hoop stress of stress of 133MPa, well within the yield strength of the HT9 at the 560 °C. . The diametral growth of pins is projected to be less than 0.6mm, hence it is unlikely there will be interference when extracting an assembly from the core and inserting it in the in-vessel thimbles and later when extracting the assemblies from the thimbles. It is noted that periodically, during operation, measurement of the pull-out forces for the assemblies will be conducted (see [Figure ApA.1]]EXPC), to ensure deformation of the ductwork remains within the acceptable range.

Operational occurrence and design/beyond design bases events can result in pool temperatures that, except for beyond design basis unprotected reactivity insertion events, approach the 560 °C temperature for relatively brief periods of time. Deterministic safety analyses conducted to date, have indicated that the maximum hot pool temperatures in all case, other than unprotected reactivity insertion events, do not exceed 564°C (Middle of life unprotected station black out event) and remain mostly under 540° C. For design basis event the temperature of the hot pool remains at 510°C. For the unprotected reactivity case, the hot pool temperatures can rise to 650 °C and remain at that temperature for several hours or days until the power excursion is terminated. At that temperature, significant additional damage, here defined as doubling the damage experienced during normal operation, would occur only if the condition were to last for a period of 90 days, during which time the condition can be rectified.

The neutron flux above 0.1MeV in the locations where the spent fuel is stored within the reactor vessel is 3 orders of magnitude lower than the flux in the core, so any additional damage would be negligible (estimate additional dpa = 0.06).

2.5 Heat Removal and Effect on Safety Margins

The storage of the spent core in vessel adds an additional source of heat to the primary and secondary system, which must be considered during all phases of operations. The greatest amount of heat added directly after shutdown is the decay heat of the core at the instant of shutdown, which is 17.3 MWth.

Refueling will take a minimum of ~20 days, during which the new core will have been installed, and the plant readied to resume power operation. When the reactor starts generating power at 100 %, the contribution of the stored core will have decayed to 0.52 MWth (20 days decay). At the same time the new core will increase its decay heat very quickly and be saturated (or very slowly increase) peaking at the discharge burnup. The combination of a stored core in vessel, with a replacement core having achieved an essentially saturated decay heat, (the new core is at the beginning of its life) with the stored core having decayed only 20 days, is what is considered in this section. Events that would occur later in the life of the replacement core are subjected to lower decay heats, since the stored core at 20 years has a total decay heat of less than 0.04Mwth.

The diverse passive heat removal system is not designed to remove the early decay heat. Two of the three trains of the Direct Reactor Auxiliary Cooling System (DRACS) are designed to remove 0.5% of the core full power rating, and the Reactor Vessel Auxiliary Cooling System (RVACS) is designed to remove another 0.2%. The longer-term highest temperatures of the fuel, cladding and coolant occur when the combined DRACS and RVACS combined heat removal capability (0.7%) matches the heat generated by the core and the stored fuel. Both systems increase their heat removal as the temperature of the cold pool increases during a transient (see [[Figures 2 through 12]]^{EXPC}, which can be found after the References). Detailed safety analyses specifically including the contribution of the decay heat of the stored cores have not yet been performed and will be done in the future, but the following can be stated based on the analyses conducted to date.

For protected events, the peak temperatures which establish the safety smallest margins occur early in the transient , before decay heat vs. decay heat removal matters. In the long term , the DRACS and RVACS have sufficient heat removal capability to cope with the decay heat from the active core plus that of the stored core and maintain temperatures well below the initial peak ones. As shown in [Figure 2 and 3]]^{EXPC}, the additional stored fuel heat has the effect of delaying the time at which the long-term temperatures peak before gradually decreasing. For the protected loss of heat sink, the heat removal matches the decay heat at approximately 4.8 hours instead of the 3 hours; and for the protected station blackout, the delay is longer (6 hrs., vs. 3.2 hrs.)

For unprotected events, those caused by reactivity insertions have the power above 100%, and the additional 0.52 MWth is a negligible addition. For such events, the reactor inherently stabilizes at a temperature which is higher than the normal operating temperature by an amount for which the negative reactivity feedback equals the amount of reactivity inserted, and analyses to date show that temperature to provide sufficient safety margins (see [[Figure 5]]^{EXPC} which shows he PTOP being virtually identical to the PSBO). For protected reactivity insertion events, the additional heat of the stored spent fuel is not an issue, because the normal heat removal pathway is available. Unprotected Loss of Heat Sink (ULOHS) results in transients for which the reactivity feedbacks do not fully terminate the fission power which remains around 0.2 percent which coincidentally is roughly the same as the heat contributed by the stored fuel (0.18%). In [Figures 9 and 11]]^{EXPC}, the initial power is assumed to be slightly above the 100% level (286.52 MWth) by the addition of the spent fuel decay heat. Alternatively , the 100% power comprises pump heating and the heat contributed by the stored fuel, so the fission power would be initially reduced by the amount of the stored fuel heat , so the residual fission power a few hours would be nearly zero. For the ULOHS, one issue to remains to be addressed in design is the cooling of the stored spent fuel. For this event, the cold pool temperatures experience a considerable rise (see Figure 10), and the flow cooling the spent fuel stored in the thimbles will be affected by the rise, whether the cold pool is connected via small orifices to the thimble or not. Results of a detailed analysis could conclude the 560°C used in section 2.4 to determine the behavior of the fuel may not be appropriate and a higher temperature should be utilized.

For the unprotected loss of flow, including station blackout (USBO), these transients exhibit their smallest margins at the beginning of the transients, as do the protected transients, but in this case the temperature of the cold pool will get a little hotter, while the hot pool will remain essentially at the same temperature. Therefore, the peak fuel, cladding and coolant temperatures will remain the same, although in the longer term, the higher temperatures will occur later and last longer.

In conclusion, the consequence of the higher decay heat (17.8 MWth) is minor for both steady state operations and transients. The margins of safety with the long-term temperatures in core and the pools being reached a period of time later than if the heat were 17.3 MWth, and the temperatures being few degrees higher. Thereafter the contribution of the stored core becomes progressively smaller. Therefore, the presence of a stored core has a small and acceptable effect on safety margins .

2.6 Decay Heat

During a normal operation, the decay heat increases quickly at the very beginning of irradiation and is saturated (or very slowly increased). The decay heat of fuel in the active core peaks at the discharge burnup, which is 17.34 MWth. The decay heat of spent fuel stored in the in-vessel storage decreases to 0.4 MW during the refueling interval of about 30 days, and it will be 0.04 MW after 20 years. Thus, the total decay heat contributed by both fuels in the active core and spent fuels in the storage will be 17.7 - 17.34 MW.

2.7 Fuel Handling Accidents

The fuel handling accidents that can occur during transfer of new fuel within the reactor vessel are the same. However, the number of times in which they can occur differs for the spent fuel. The IVTM handles the spent fuel once when extracting it from the core and transferring to the storage location; and a second time when it transfers it from the storage location to the transfer port to be picked up by the FUM, removed from the reactor vessel and placed in the transportation cask. During the first transfer, the spent fuel assembly will travel over portions of the core , and an accidental drop of the in-transfer assembly can cause some self-damage and damage to the remaining core assemblies. [This could also occur during the transfer of a core assembly directly to the transfer port for extraction and transportation to an ex-vessel storage tank.] The second transfer would not require passing over the spent or the new core but will require a transfer over some of the already stored spent fuel assemblies. A drop during this transfer would basically have similar consequences to the first transfer, with the sole exception that new fuel would not be involved. Dropping an assembly in vessel will require an extensive outage to retrieve the assembly as illustrated in the case of the Joyo plant in Japan . From this standpoint external storage would seem preferable. However, externals storage still entails retrieval from the core to the transfer port, and then from the transfer port to the ex-vessel storage, lowering to the ex-vessel storage, and then from the ex-vessel storage to the transportation cask instead of from the transfer port directly to the transportation cask. So, instead of handling the spent fuel twice from its storage in vessel to the transportation cask via the transfer port and FUM, the fuel is handled three times, from the transfer port to the ex-vessel storage tank, storing in the ex-vessel tank, retrieved from the ex-vessel tank and placing in the transportation cask. In summary in-vessel spent fuel storage is susceptible to in-vessel or in reactor building fuel accidents, whereas ex-vessel storage also adds susceptibility to accidents outside the reactor building, which acts as the functional containment.

In terms of fuel handling accidents that are not assembly drops; but can results in fuel damage as a result of overheating if an assembly were to become stuck during transfer and be deprived of cooling, storage in vessel is preferable because transfers outside the reactor sodium pool occur when the fuel has decayed to levels that do not require cooling.

2.8 Remaining Issues to be Addressed

As already mentioned in section 2.5 the effect on fuel behavior during a ULOHS event remains to be determined once a decision on design of the storage thimble is made. The design of the thimbles will depend on whether gaps between the thimble ID and the assembly promote sufficient natural circulation of the hot pool coolant under the influence of the assembly decay heat, or small orifices (weep holes) need to allow cold pool coolant to aid the assembly cooling.

If the latter, depending on the design of the thimble supports a small fraction of the coolant could bypass the core, and that effect remains to be established.

3.0 REGULATORY BACKGROUND AND PRECEDENTS

No NRC or other regulatory bodies or IAEA regulations have been identified that specifically address storage within the reactor vessel. Regulations and guidance are provided for storage in general and those are used in the design, whether in or out of vessel. There have been several (as a fraction of the worldwide sodium cooled reactors) precedents for storing spent fuel in-vessel. Appendix C lists the world sodium cooled reactors by kind (experimental, demonstration or prototype, and commercial), type (loop or pool) , and with their refueling and spent fuels storage methods.

4.ACRONYMS AND DEFINITIONS

4.1 Acronyms

ANL	Argonne National Laboratory
EAF	Energy availability factor
EBR-II	Experimental Breeder Reactor II
DRACS	Direct A Reactor Auxiliary Cooling System
FUM	Fuel Unloading Machine
GWh	Gigawatt-hour
IAEA	International Atomic Energy Agency
IVTM	In-Vessel Transfer Machine
LF	Load Factor
MWh	Megawatt-hour
OF	Operation Factor

PEL	Total planned energy losses over the specified period
PRIS	IAEA Power Reactor Information System
REG	Reference Energy Generation over a specified period
RVACS	Reactor Vessel Auxiliary Cooling System
UCF	Unit Capability Factor
UCL	Unplanned Capability Loss Factor
UEL	Total unplanned energy losses over the specified period
XEL	External Energy Loss

4.2 Definitions

The following definitions are obtained from the IAEA Power Reactor Information System (PRIS) 0.

Capacity Factor (CF)	The actual energy output of an electricity-generating device divided by the energy output that would be produced if it operated at its rated power output (Reference Unit Power) for the entire year. Generally expressed as percentage. In PRIS a term Load Factor (LF) is used for CF.
Energy Availability Factor (EAF)	The energy availability factor over a specified period, is the ratio of the energy that the available capacity could have produced during this period, to the energy that the Reference Unit Power could have produced during the same period.
Load Factor (LF)	Load Factor, also called Capacity Factor, for a given period, is the ratio of the energy which the power reactor unit has produced over that period divided by the energy it would have produced at its reference power capacity over that period.
Operation Factor (OF)	Operation factor is defined as the ratio of the number of hours the unit was on-line to the total number of hours in the reference period, expressed as a percentage. It is a measure of the unit time availability on the grid and does not depend on the operating power level. On-line hours are the total clock hours in the reference period during which the unit operated with breakers closed to the unit bus. Reference period hours are the total number of hours in the pre-defined calendar time.
Reference Energy Generation (REG)	Reference energy generation (MWh or GWh) for the period is the net electricity output that would be produced if a reactor unit is operated at its rated power output for the entire period.
Reference Unit Power	The reference unit power expressed in units of megawatt (electrical) is the maximum (electrical) power that could be maintained continuously throughout a prolonged period of operation under reference ambient conditions. The power value is measured at the unit outlet terminals, i.e., after deducting the power taken by unit

	auxiliaries and the losses in the transformers that are considered integral parts of the unit. The reference unit power is expected to remain constant unless following design changes, or a new permanent authorization, the management decides to amend the original value.
Unit Capability Factor (UCF)	Unit capability factor is defined as the ratio of the available energy generation over a given time period to the reference energy generation over the same time period, expressed as a percentage.
Unplanned Capability Loss (UCL)	The ratio of the unplanned energy losses during a given period of time, to the reference energy generation, expressed as a percentage. Unplanned energy loss is energy that was not produced during the period because of unplanned shutdowns, outage extensions, or unplanned load reductions due to causes under plant management control. Energy loss is considered to be unplanned if it is not scheduled at least four weeks in advance.

5.REFERENCES

[1] R. C. Iotti, "Calculation to compare Lanthanides generated layer thickness in experiments carried out at the EBR II and FFTF with predictions of reference 3 and consequences to the ARC 100 long term possible weakening of the cladding" July 2-5, 2021

[2] BISON Fuel performance Assessment of the Reference ARC-100 Fuel Design During Normal Operation, IN?/RPT-23-71508

[3] K. Inagaki, K, Nakamura, T. Ogata, and T.UWABA, " Chemical Interaction in Metal Fuel for FBR", Vol2, No 2, p 149-157, (2013, Doi:10.3327/taesj.J12.007

[4] T. Sumner and A. Moisseytsev, '286 MW_{th} Core ARC-100 Safety Analysis" ANL/NE-ARC/ARC20-05, Rev 1, May31, 2022

[5] T. Fei, T.K. Kim, and C. Grandy," Preliminary Assessment if the Flux and Cross Sections for the ARC-100 Components near the Reactor Core, (rev 00)' ;ARC-RPS-001, December 7, 2021

[6] IAEA Technical Report Series No. 428, *The Power Reactor Information System (PRIS) and its Extension to Non-electrical Applications, Decommissioning and Delayed Projects Information*, Vienna, 2005.

[7] ARC-PRD-001, Rev. B, *ARC-100 Project Requirements Document*.

[8] E. A. Hoffman, T. Fei, and T. K. Kim, *286 MW_{th} ARC-100 Core Design Report, Rev. 05*, Nuclear Science and Engineering Division, Argonne National Laboratory, November 4, 2019.

[9]" Bonneville Power Administration, BPA Balancing Authority Load and Total Wind, Hydro, Fossil/Biomass, and Nuclear Generation, Near-Real-Time". *transmission.bpa.gov*. January 6–13, 2017.

[[Figure 2. BOL PLHOS – Power, Long Term (from Figure 3.32 of Ref. 4]]EXPC

[[Figure 3. BOL PSBO – Power (from Figure 3.7 of Ref. 4]]EXPC

[[Figure 4. BOL PSBO – Peak Core Temperatures, Short Term (from Figure 3.3 of Ref. 4]]EXPC

[[Figure 5. BOL PTOP– Power, Long Term (from Figure 3.21 of Ref. 4]]EXPC

[[Figure 6. BOL PTOP– Peak Core Temperatures, Short Term (from Figure 3.18 of Ref. 4]]EXPC

[[Figure 7. BOL UTOP (Unexpanded Fresh Fuel)- Reactivity Feedbacks (from Figure 4.23 of Ref. 4]]EXPC

[[Figure 8. BOL UTOP (Unexpanded Fresh Fuel)- Peak In-Core Temp's (from Figure 4.24 of Ref. 4]]EXPC

[[Figure 9. BOL ULHOS– Power, Long Term (from Figure 4.31 of Ref. 4]]EXPC

[[Figure 10. BOL ULHOS– Hot and Cold Pool Temperatures (from Figure 4.32 of Ref. 4]]EXPC

[[Figure 11. BOL USBO – Power, Long Term (from Figure 4.6 of Ref. 4]]EXPC

[[Figure 12. USBO Power (left)– Hot and Cold Pool Temperatures right (from Figure 4.32 of Ref. 4]]EXPC

APPENDIX A

ARC 100 CAPACITY FACTOR

This Appendix A describes the basis for the fuel cycle length and capacity factor estimated for the ARC-100 Facility

Basis for Unit Capability Factor

As defined by the IAEA 0, the unit capability factor for electricity generation is a measure (expressed as a percentage) of how often an electricity generator operates during a specific period of time using a ratio of the actual output to the maximum possible output during that time period:

$$\text{Unit Capacity factor (\%)} = \frac{(\mathrm{REG} - \mathrm{PEL} - \mathrm{UEL})}{\mathrm{REG}} \times 100$$

where,

REG is the reference energy generation: The net electrical energy (MWh), supplied by a unit continuously operated at the Reference Unit Power for the duration of the entire operation period, i.e., REG for ARC-100 is 100 times the number of hours.

PEL is the planned energy loss: The energy (MWh) that was not supplied during the period because of planned shutdowns or load reductions due to causes under plant management control. Energy losses are considered to be planned if they are scheduled at least four weeks in advance.

UEL is the unplanned energy loss: The energy (MWh) that was not supplied during the period because of unplanned shutdowns, outage extensions or load reductions due to causes under plant management control. Energy losses are considered to be unplanned if they are not scheduled at least four weeks in advance.

The unit capability factor reflects the effectiveness of plant programs and practices in maximizing available electrical generation and provides an overall indication of how well a plant is operated and maintained.

The IAEA 0 also defines the term Load Factor to be the same as Capacity Factor:

$$\mathrm{LF}\ (\%) = \frac{\mathrm{EG}}{\mathrm{REG}} \times 100$$

where,

EG is the electrical energy: The net electrical energy (MWh) supplied during the reference period, as measured at the unit outlet terminals after deducting the electrical energy taken by unit auxiliaries and the losses in transformers that considered to be integral parts of the unit.

ARC-PRD-001 0 has the following requirements related to the unit capacity factor or load factor:

[a]	The ARC-100 plant shall be designed to achieve a target lifetime capacity factor of at least 90%.

	Rationale: This lifetime target of greater than 90% is based on accounting for maintenance outages, two refueling outages, forced outages and assumptions regarding low electricity demand.
[b]	The ARC-100 plant shall be designed to achieve a target annual capacity factor of greater than 90%, with the exception of the First of a Kind facility .
	Rationale: This annual target of greater than 90% is based on accounting for maintenance outages, forced outages and assumptions regarding low electricity demand, and support achievement of the requirrment above, but for a FOAK, during the first 20 years, a number of assemblies wil be extraccted to verify the fuel performance. This will require additional planned shutdowns
[c]	The ARC-100 plant shall be designed to target less than one unplanned forced outage, either by automatic shutdown due to a reactor trip or manual shutdown, per plant year.
	Rationale: This requirement supports achieving the safety objective of minimizing the frequency on demand of the safety systems and the target annual capacity factor.

Examples of reasons for having a unit capacity factor less than 100% include:

Reduced output for primary or secondary frequency control for the electrical grid: Nuclear power plants include an automatic operating mode to respond to grid frequency changes. When the reactor is responding to primary or secondary grid frequency changes, the reactor power is typically modulated by ±2-3% for primary frequency control and ±3-5% for secondary frequency control. If the reactor power is restricted by its operational limits and conditions to never exceed 100% power, a nominal power setpoint of less than 100% power, such as using 99%, is used with ±2% variation for frequency control. This results in the plant capacity factor being less than 100%. Some reactors are allowed to operate with power modulated around 100% power. In these cases, the nominal power setpoint is 100% power, where a ±2% variation for frequency control has resulted in the capacity factor exceeding 100% for a year. This contributes to the PEL.

Load following: When a reactor is used for load following with the reactor power being maneuvered in a range, such as 30-100%, to match daily electrical power demands, the capacity factor is reduced. A requirement in ARC-PRD-001 0 specifies that the ARC-100 plant must be capable of daily load cycling from 25-100% power based on grid power demands, with a rate of change of the electric output of 3-5% of rated power per minute. This requirement is consistent with a typical utility requirement to support load following with high ride-out (load rejection) capability to support efficient and effective grid operation in all market conditions. This contributes to the PEL. It is noted that load following could result in the plant capacity factor falling below the 90% requirements above. Because the amount of time during which the ARC 100 will be in a load following mode cannot be ascertained at this time, an assumption is made that the load following is similar to that experienced by the Bonneville Power Administration [9], in which the load provided by the plant on a weekly is less than 100% 5 times per week, each time for 8 hours at an average power reduction of 4.375% . This corresponds to an outage of 120 days in 20 years. In reality if load following is going to be to be a frequent event and with deeper power reductions, this assumption will be incorrect, and the capacity factor may be significantly reduced below the estimated 90%

Equipment failures: Minor equipment failures can cause a reactor to operate at reduced power until a scheduled maintenance outage occurs. This contributes to the UEL.

Maintenance outages: The period of time when a reactor is shut down for routine maintenance reduces the plant capacity factor for the year in which it occurs. For example, a 30-day maintenance outage reduces the plant capacity factor by about 8% in the year in which it occurs. This contributes to PEL.

Planned Outages: Periods of time during which the plant is shutdown to conduct measurements of pull-out forces in core assemblies as part of verifying core deformations over time are within the acceptable limits, or required in service inspections are conducted, or refurbishments of power conversion (e.g., turbine generator, compressors) equipment need to take place. This contributes to PEL.

Forced outages: Unplanned equipment failures or external events can cause a reactor to be shut down for a period of time. There is operating experience where nuclear power plants have been forced to operate at reduced power or to shut down for periods of time to satisfy environmental limits for temperatures in the ultimate heat sink (e.g., water temperature in a discharge zone in a water body used as the ultimate heat sink). This contributes to the UEL.

Projected Performance Factors

Section A.1 provides projections for the annual plant capacity factors for the ARC-100 plant, the results of which are summarized in the table below.

Capacity Factor	Lead Unit (FOAK)		Subsequent Units	
	Seq. Maint.	Overlap Main.	Seq. Maint.	Overlap Main.
Cumulative 1st Cycle (1 to 20 years)	90.5%	92%.0	91.6%	93.0%

ARC-PRD-001 0 has the following requirements related to the fuel cycle length:

	The nominal fuel design life shall sustain a core thermal power of 286 MW_{th} for a minimum of 18 full-power years to accommodate refueling intervals of 20 years.
	Rationale: The fuel inventory in the core is required to sustain generating 286 MW_{th} at greater than 90% capacity factor for a minimum of 18 years.

The basis for requirement for the fuel cycle length is:

- ARC-PRD-001 0 specifies a net electrical output of 100 MWe. Based on the thermal to electrical efficiency of the power conversion system, a core thermal power of 286 MW_{th} is needed to achieve the net electrical output of 100 MWe.
- The strategic objective is to have a nominal refueling interval of 20 years. The ANL report, *286 MW_{th} ARC-100 Core Design Report* **Error! Reference source not found.**, describes the basis for achieving a refueling interval of 20 years, assuming an average capacity factor of 90% over the 20-year interval. The average daily burnup is ~0.97 MWD/g heavy metal (i.e., uranium and plutonium).
  - Several design constraints are imposed in the ANL report **Error! Reference source not found.** on the design of the core, including:
  - Core barrel limited to 3 m diameter to allow transportation to local and small grid areas from a pre-licensed factory,

- Maximum burnup reactivity swing was limited to have a sufficient shutdown margin with a reasonable number of control assemblies and provide passive safety feature (i.e., inherent reactivity feedback), and
- Fuel design to be within operating experience for U-10%Zr fuel.

For periods of time when the plant capacity factor falls below 90%, the inventory of fissile and fissionable material in the core would enable the reactor to operate for more than 20 years before refueling is required. The periods of time when the plant capacity factor exceeds 90% will result in reducing the time to refuel the core to less than 20 years.

A.1. Projected Plant Capacity Factors for ARC-100 Plant

Table A-1 shows the projected durations for planned maintenance activities for the ARC-100 standard plant in the years after reaching 100% power and being connected to the supply electricity to the grid, as shown in Figure A.1. These durations and the timing are assumed to be applied to each of the three 20-year operating cycles for the 60-year design life of the plant. Note that Table A-1 does not include planned outages for refueling, nor outages for extraction of FOAK assemblies, planned to demonstrate performance of the fuel, nor periodic hot shutdowns of the FOAK for measuring assemblies pull out forces.

[[Table A-1: Projected Durations and Timing for Planned Maintenance Activities]]P

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Notes:

1. **Control rods absorber replacements**: this projection assumes that the safety rods do not need replacement, and the control rods are NOT stored in-vessel),because the number of available spaces may not be sufficient, but are removed from the vessel an transported to either the radwaste building or rods projected the durations to be 2 days for cooling and preparation of IVTM and FUM, 22.3 hours for removal and replacement, 1 day to restart for a total of ~4. days per replacement. If at the first replacement, there is space for in vessel storage, the first replacement will requires less than half day (9.6 hours) and the outage will be only 3.5 days
2. **In-Service Inspection (ISI)**: projected the durations to be shutdown with 12 hours cooling, ISI activity 3 days, restart 1 day for a total of 4.5 days per ISI.
3. **Major Overhaul of Balance of Plant**: projected the durations for the Brayton cycle turbine generator to be 35 days for each major maintenance outage (this period is based on typical outage of refurbishment of steam and combustion turbines, as there is no data for supercritical CO_2 turbine. Table A-2 shows the projected durations for planned refueling activities for the ARC-100 standard plant in the years after reaching 100% power and being connected to supply electricity to the grid from a time and motion study of two spent fuel storage options analyzed in Appendix B.

[[Table A-2: Expected Durations and Timing for Planned Refueling Activities]]^P

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The regulatory requirements for an annual report on fuel monitoring and inspection impacts the capacity factor whenever an outage is needed to replace a fuel assembly before reaching the end-of-life burnup for the core.

Since visual inspections of the ARC-100 spent fuel assemblies is not possible at the plant site, the fuel monitoring and inspection program for the first 20-year operating cycle for the FOAK ARC-100 plant is projected to rely on periodically sending an irradiated fuel assembly to an external laboratory for post-irradiation examination. If the performance of the fuel is maintained within its design criteria, it may be possible to justify not requiring periodic removal of a fuel assembly for inspection during the 2nd and 3rd 20-year operating cycles for the lead plant and during the three operating cycles for subsequent plants. In a performance-based approach, a replacement of a fuel assembly during an operating cycle would only take place in the event that a fuel defect causes fission products leaking into the sodium coolant to reach the limit defined in the operational limits and conditions, which for the ARC 100 is 0.1% failed fuel. A fuel assembly from each whole core refueling could also be sent for post-irradiation examination without impacting the capacity factor.

Using the information in Tables A-1 and A-2, Table A-3 summarizes projected outages during the 1st 20-year operating cycle for the lead unit using the following assumptions:

- In years where there are multiple outage activities, the Sequential column projects a total outage duration where the durations for each type of outage activity are assumed to occur sequentially to give a very conservative impact on the capacity factor.
- In years where there are multiple outage activities, the Overlapped column projects a total outage duration where the durations are assumed to occur in parallel to give a best estimate impact on the capacity factor.

[[Table A-3: Lead Unit Projected Outages For 1st 20-Year Operating Cycle]]P

Table A-3 does not include unplanned outages or unplanned maintenance, nor the less than 100 percent power days due to frequency regulation and load following. However, it is used to illustrate how scheduling work to be coincident has a significant effect on the capacity factor. The difference between the Sequential work and the Overlapping work is about 100 days, or about 1.4% in capacity factor. Moreover, is shows the Capacity Factor determined in Figure ApA-1 by a combination of mostly sequential work assumptions is conservative and may compensate for the uncertainty in the effect of particularly the load following.

[[Figure ApA.1 Capacity Factors Estimated for the FOAK and NOAK ARC 100 Facilities]][P]

APPENDIX B

ARC-100 CORE RELOADING EVALUATION

ARC-100 FUEL HANDLING TIME EVALUATION

ASSUMING IN-VESSEL SPENT FUEL STORAGE

First and Subsequent Core Reload

Assumptions:

<u>First Fueling</u>

Driver fuel positions in core are filled with dummy assemblies, the in-vessel storage positions are empty and new fuel is brought in to replace the dummy assemblies. Since the time is shorter for this operation than the operation involved in the first refueling, and the reactor is not yet operating, this scenario is not examined in this time and motion study. However it is noted that to compare the in vessel spent fuel storage to the ex-vessel fuel storage on the same basis, one should assume that the in vessel storage position at the first refueling are empty (i.e. have been disposed of in a location within the reactor building or the radwaste building- which would be the same assumption as is made in the ex-vessel storage time and motion study, in which it is assumed that the dummies in the core are extracted from the core one at a time, and replaced by one driver assembly at the time and disposed of before power operation begins).In this time and motion study we chose instead to dispose of the dummies by storing them in the empty in vessel storage location, which then requires extra motions to remove them at the first refueling . The reason for this choice is the advantage (benefit) derived by locating the dummy assemblies in the in-vessel storage location where they can provide shielding to further reduce the activation of the secondary sodium the DRACS NaK and the air in the RVACS. The disadvantage with pacing them in the in-vessel storage location is their activation and the extra time needed to remove them from the reactor vessel upon refueling.

<u>First Fueling and Subsequent Refueling</u>

Reactor Shutdown with primary sodium cooled to fuel handling temperature is assumed to require 2 days.

- Refueling temperature is achieved in 10-12 hours and installation of readiness of IVTM and readiness of FUM takes another 36 hours.

For this option it is not necessary to wait 15 days for the spent fuel to decay to levels acceptable by the FUM, because the spent fuel remains in the reactor vessel until the next refueling.

For the first refueling, the in-vessel storage locations (other than the central one below the FUM port) will be filled with dummy fuel assemblies which must be removed, one by one, to make space for the spent fuel. This is accomplished by first using the IVTM to transfer a dummy assembly to the central in vessel storage position, where the FUM will take it and convey it out of the reactor vessel and transfer it to a cask located within the reactor building for ultimate transport out to the Radwaste Building for remote storage. For subsequent refueling the in-vessel storage locations will be filled with spent fuel from the prior cores.

Rotating plug and seals ready for fuel handling, all other system conditions ready for in-vessel fuel handling

Fuel Handling Step	Time To Complete (minutes)	Elapsed Time (minutes)	Comments
0. Shutdown and ready IVTM and FUM	2 days		The two days consist of 12 hours to reduce temperature and 36 hours to prepare the IVTM and FUM. The two days are added at the end of the time and motion study. **Note:** Would utilize the two-day period to load the new fuel racks (50% of core capacity) and bring in the initial dry spent fuel cask or dummy cask
1. IVTM travels to in vessel storage location, engages dummy assembly (1st fueling) or spent assembly (subsequent refueling), and raises it to transport elevation	10	10	**Note:** Not part of refueling, but when using the IVTM to measure pull out force, the time needed to do so will likely increase from 10 to 18 hours. But in addition, 12 hours will be required to reduce the temperature and insert the IVTM, which is normally out of the vessel. Another 18 hours to insert the IVTM. With the control elements decoupled, the rotating plug will rotate to expose the assemblies. The forces measured during the 18 hours. The IVTM will be withdrawn from the vessel, the control elements recoupled, and the temperature increased to resume power operation (another 12 hours) These operations are expected to require 36 hours. Doing the force measurement will need a total of 96 hours.
2A. IVTM moves dummy or spent assembly from in-vessel storage to central storage location, lowers assembly into central storage location, releases assembly and	15	25	

Fuel Handling Step	Time To Complete (minutes)	Elapsed Time (minutes)	Comments
rotates to parked position			
2B. FUM and fuel transfer ports open and gripper is lowered into central storage position	10?	35	
2C. . FUM and fuel transfer ports open and assembly (dummy or spent fuel) is gripped	5?	40	
2D. FUM gripper is retracted from reactor vessel, holds for Na drain	30	70	
2E. 8. FUM gripper fully raised into FUM, transfer port closes, FUM port closes	5	75	
2F. FUM travels from top of reactor head location to wash area in reactor building where assembly (dummy or spent) is washed.	15	90	FUM travel is 30 ft totally within reactor Building, • FUM Travel – 5 min • Connection to wash station -5 min • Lowering, releasing element, raising gripper, closing connection -5 min
2G. Assembly is washed	20?	110	Includes recovering element with FUM and closing connection
2H. FUM proceeds to dry cask and lowers dummy or spent fuel in cask.	15	125	See wash station - travel distance from wash station to dry cask is ~15 feet (furthest slot) Note. Since transport cask will hold more than 33 , but certainly not the entire core, it is assumed that once the transfer casks house 33 assemblies, the operations with the FUM will suspend until the cask is transported to the onsite dry cask storage facility, and the cask is returned to the reactor and readied to receive additional spent fuel assemblies. The time for this operation is addressed in
3 FUM travels to fresh fuel storage rack	5	130	
4. FUM gripper engages fresh assembly and raises	5	135	

Fuel Handling Step	Time To Complete (minutes)	Elapsed Time (minutes)	Comments
it into FUM. FUM transfer port closes. Argon purge and subassembly preheat initiated			
5. FUM travels to fuel transfer port. FUM port is sealed to storage vessel port and air gap is purged with argon	30	165	Time assumes assembly preheated at end of travel.
5A IVTM goes from central storage location or parking position to core position, engages spent assembly (1st or subsequent refueling), and raises it to transport elevation	10	165	In parallel with Step 5
5B. IVTM moves spent assembly from core location to in-vessel storage location previously vacated, releases assembly and rotates to parked position or position near central storage location	15	165	In parallel with Step 5
6. FUM and fuel transfer ports open and fresh assembly is lowered into central storage position	13	178	If assembly is not at proper temperature, additional time is required for preheat.
7. FUM gripper is retracted from reactor vessel, holds for Na drain	15	193	Since the gripper does not hold a fuel assembly the time required to Na drain is less than the time needed when an assembly is suspended within the FUM
8. FUM gripper fully raised into FUM, transfer port closes, FUM port closes	5	198	
STATUS: At this point, there is a fresh assembly in the central transfer position and an open core location. A spent driver assembly is in in-vessel storage.			

Fuel Handling Step	Time To Complete (minutes)	Elapsed Time (minutes)	Comments
9. IVTM travels to central transfer position, engages fresh assembly, and raises to transport elevation	10	208	
10. IVTM moves fresh assembly to open core location, lowers assembly into core, releases assembly, raises and rotates to parked position or to next core position. STATUS: One core assembly replacement complete at this point. IVTM is ready to remove next assembly from in vessel storage (Step 1.).	15	223	NOTE: Steps 1 through 10 are critical path
11. Repeat for a total of 99 core assemblies.		99 transfers, 223 minutes per transfer: 22,077minutes (368hours) (15.3 days) plus 24hours (1.0 days) plus 2days before start movement of fuel yields total outage of about **18.3days**	1. Time required for full core reload with fresh fuel; assuming sufficient in-vessel spent fuel storage for complete core 2. RB only holds ½ core of new fuel. During refueling, the prior operations must stop to bring in the second half of new fuel assemblies, estimate 12 hours. 3. Transport cask does not hold entire core, assume will have to be replaced twice, estimate 12 hours total (assume FUM would be stationary while crane is being used) To avoid the transportation cask being on the critical path of refueling ,two transportation casks are assumed to be available, so that one can be used to transport the spent fuel to the dry-cask storage facility and unload the spent fuel , while the second one is moved to the reactor building and readied to receive spent fuel. The movement of the loaded transportation cask out of the reactor building to make space for the second cask movement to

Fuel Handling Step	Time To Complete (minutes)	Elapsed Time (minutes)	Comments
			the FUM location should take less than 12 hours. 4. Tasks 2 and 3 above = 24 hours (1 Day)
Status: Core is reloaded with fresh fuel, in-vessel storage contains complete core of spent fuel assemblies. Prior to next core reload, spent fuel assemblies must be removed.		same as above.	

ARC-100 FUEL HANDLING TIME EVALUATION

ASSUMING NO IN-VESSEL SPENT FUEL STORAGE

FIRST (and Subsequent) CORE RELOAD

Assumptions:

First fueling assumes driver fuel position in core are filled with dummy assemblies, the in-vessel storage positions are empty and new fuel is brought in to replace the dummy assemblies. Since the time is shorter for this operation than the operation involved in the first refueling, and the reactor is not yet operating, this scenario is not examined in this time and motion study However a reasonable estimate of the time it will take to replace the dummies in the core with the driver assemblies is estimated in the initial steps of the time and motion study for in-vessel spent fuel storage . It is noted that it is tacitly assumed that the dummies are stored in a location either in the reactor building or the adjacent radwaste building, The dummy assemblies are non-radioactive.

Reactor Shutdown, primary sodium cooled down to fuel handling temperature (can be done in 12 hours but assume one day).

IVTM Installed and in parked position, rotating plug and seals ready for fuel handling (cannot be done until reactor is at refueling temperatures but is not part of time and motion study) Note that for this option , the initial fueling is not considered, since the reactor is not yet operating. Therefore, the initial replacement of the core driver fuel positions initially filled with dummy assemblies to new drivel fuel assemblies is not considered as part of this time and motion study, and the core is assumed to be full of spent fuel assemblies that have to be replaced.

FUM is sealed to transfer port, ready to receive assembly (same as directly above, but conducted in parallel with set up of IVTM and also not part of this time and motion study)

All other system conditions ready for in-vessel fuel handling

The time and motion study, for this option, considers the first refueling only, and it assumes a that at the end of this refueling the spent fuel storage tank located in the auxiliary building is fuel (including control elements that have been replaced at least twice before the refueling. The second refueling is identical, because it is assumed that before that occurs, the spent fuel storage tank is emptied, by transferring the stored fuel to outdoor dry storage casks.

It is also assumed that the last fuel handling is the removal of all of the fuel control elements a reflector and shield elements from the core, with the driver fuel being stored in the spent fuel storage tank in the auxiliary building, and the other assemblies disposed of as intermediate level waste.

Because the driver fuel is transferred out of the vessel, a period of time must elapse from shutdown to the first transfer, in order for the decay heat of the assembly to be within the capabilities of the FUM to handle. This period of time has been calculated (on a preliminary basis) to be a minimum of 15 days (comparable to the EBR II experience. In the table below this is considered the zero step

Fuel Handling Step Comments	Time To Complete (minutes)	Elapsed Time (minutes)	Notes
0. Allow spent fuel to decay to level that FUM can handle	15 days	15 days	The 15 days are added to the total of the actual fuel handling at the end of this time and motion study. The 15 days include the10- 12 hours necessary to lower the pool temperature to 200 C or less and the time to get the FUM and IVTM ready
1. IVTM travels to core location, engages spent assembly, and raises it to transport elevation	10	10	
2. IVTM moves spent assembly to fuel transfer position, lowers assembly into central transfer position, releases assembly and rotates to parked position	15	25	
3. FUM and fuel transfer ports open and gripper is lowered into central storage position FUM gripper engages assembly	15	40	FUM gripper lowered and prongs engage assembly
4. FUM gripper raises assembly and holds for Na drain	30	70	

5. FUM gripper raises assembly fully into FUM; transfer port closes	5	75	
6. FUM travels to fuel storage tank in Auxiliary Building (~90 ft) through airlock and positioned over storage tank target location. FUM port is sealed to storage vessel port and air gap is purged	35	110	Previously or concurrently, rotating plug on storage vessel is rotated to proper position for receiving assembly. FUM travel time includes airlock operation time
7. Fuel storage vessel port opens and FUM gripper lowers assembly into storage tank and releases it	15	125	
8. FUM gripper is retracted from storage tank, holds for Na drain	15	140	Since the gripper does not hold a fuel assembly the time required to Na drain is less than the time needed when an assembly is suspended within the FUM
9. FUM gripper fully raised into FUM, transfer port closes, FUM port closes	5	145	
10. FUM travels to fuel storage rack (~ 40 ft) and picks up new fuel assembly, raises assembly into FUM, closes FUM port, and begins heating assembly	20	165	
11. FUM travels to fuel transfer port through airlock (~130 ft)	45	210	FUM preheats assembly during transport
12. FUM is positioned over transfer port, FUM port is sealed to transfer port, air gap is purged	5	215	
13. FUM and transfer ports are opened, and assembly is lowered to central transfer position	13	228	If assembly is not at proper temperature, additional time is required for preheat.
14. FUM gripper raised into FUM, holds for sodium drain	15	243	Since the gripper does not hold a fuel assembly the time required to Na drain is less than the time needed when an

			assembly is suspended within the FUM
15. IVTM rotates to fuel transfer position, engages assembly, raises assembly to transport elevation	14	258	
16. IVTM rotates to empty core position, inserts new assembly into core	10	268	
17. Repeat Steps 1-16 for 98 more core assemblies. Status: the core now has fresh fuel, and the spent fuel storage vessel contains all of the spent fuel assemblies from the core.		99 round trip transfers: 99x268= 26,532 (442 hours) (18.4 days)	Time required for full core reload with fresh fuel; no in-vessel spent fuel storage
Total time for this refueling activity		18.4+15=33.4 days	
18. The spent fuel assemblies will be removed from the spent fuel storage vessel and transported to long term storage facility after sufficient decay time has elapsed.	TBD	TBD	Time required to remove spent fuel from storage vessel and transport to long-term storage is not plant-limiting, if it is done in between refueling outages and is not included in this evaluation. However, it will have to be done, whereas of the in-vessel time and motion study, this will already be done. The advantage of the ex-vessel storage is that only one transportation cask will be needed, whereas for the in-vessel storage two have to be available if movement to the dry storage facility is to be kept out of the critical path.

APPENDIX C

REFUELING AND SPENT FUEL STORAGE METHODS

Experimental Fast Reactors	
Rapsodie (France)	2 RP and 2 VM
KNK-II (Germany)	
FBTR (India)	2RP and 2VM
PECataM	1 RP PM under VH
Joyo (Japan)	VM in 2 RP **IVS (32/67) [C4]**
DFR (UK)	VM in2 RP
BOR-60 (Russian Federation)	VM in 2 RP
EBR II (USA)	VM in 2 RP and trasnfer arm
FERMI 1 (USA)	VM with fixed exit, RP with offset mechanism
FFTF (USA)	3 VM eachin 1 RP
BR-JO (Russian Federation)	2 RP and 1 VM
CEFR (China)	VM in 2 RPs. **IVS (?/?)**
Demonstration or Prototype Reactors	
Phenix (France)	Fixed offset arm in 1 RP, **IVS(?/?)**
SNR-300 (Germany)	VM in 3 RP
PFBR (india)	Fixed offset arm in 2 RP, **IVS(156/181) [C3]**
MONJU (Japan)	Fixed offset arm in 1RP, **IVS (89/196) [C4]**
PFR (UK)	PM in 1 RP**, IVS (?/?)**
CRBR (USA)	VM in 3 RP
BN-350 K(azakhstan)	VM in 2 RP; **IVS (41/199)[C4]**
BN-600 (Russin Federation)	VM in 2 RP; **IVS (126/297)[C4]**
ALMR (USA)	2 PM in 2 RP
KALIMER-150 (Republic of Korea)	PM, RP, **IVS (114/336) [C2]**
SBVR-75/100 (Russian Federation	VM
BREST-OD-300 (Russian Federation)	2RP+VM+rotating mechanism+ horizontal trasfer mechanism
Commercial Size Reactors	
Super-Phenix-1 (France)	2 VM in 2 RP
Super-Phenix-2 (France)	2 VM in 2 RP
SNR 2 (Germany)	Underhead to trasfer position
DFBR (Japan)	IVM in 2 RP
BN-1600 (Russian Federation)	VM in 3RP, **IVS (?/?)**
BN-800 (Russian Federation)	VM in 3RP, **IVS (192/565) [C1]**
BREST-1200 (Russian Federation)	2RP, VM,**(to be defined)**
JSFR-1500) (Japan)	1PM in 1RP

RP- Rotating Plug
VM- Vertical mechanism (direct lift)
VH -Vessel head
FM -F ixed-arm mechanism
PM -Pantograph mechanis m
IVS -In vessel storage (no. stored in vessel/total number) [Reference]

[C1]https://media.superevent.com/documents/20170620/11795dbfabe998cf38da0ea16b6c3181/fr17-405.pdf
[C2]https://www.kns.org/files/pre_paper/14/656%EA%B9%80%EC%84%9D%ED%9B%88.pdf
[C3] https://fissilematerials.org/library/igcar04.pdf

[C4] https://digital.library.unt.edu/ark:/67531/metadc868909/m2/1/high_res_d/4032678.pdf

ANNEX 1

APPROXIMATE CRITICALITY CALCULATION FOR PROPOSED SPENT FUEL IN VESSEL STORAGE CONFIGURATION

This calculation is not to determine the specific neutron multiplication factor for the spent fuel configuration proposed for in-vessel storage. It is done to determine whether that configuration could go critical. A detailed critical analysis, performed with the appropriate computer code, should be done once the configuration details are firmed up, and prior to finalizing the actual configuration with specific materials.

From this calculation, documented below, it is concluded that the proposed configuration will be subcritical.

In a configuration of masses that include fissionable and fissile material, criticality is achieved when the rate of neutron production is equal to the rate of neutron losses, wherein the latter include both neutron absorption and neutron leakage. Geometric buckling is a measure of neutron leakage, while material buckling is a measure of neutron production minus absorption.

Both buckling terms are derived from the diffusion equation [2]

$-D\,\nabla^2\Phi + \Sigma_a\,\Phi = 1/\kappa\ \nu\,\Sigma_f\,\Phi$, where k is the criticality eigenvalue, Σ_f and Σ_a are the macroscopic fission and absorption cross sections respectively, and from diffusion theory, the diffusion coefficient, *D*, is defined as $D=1/3\,\Sigma_{tr}$. *(Eq. 5.15.and 5.16 of Ref 2)* The diffusion length, L, is defined as $L= (D/\,\Sigma_a)^{1/2}$ *(Eq. 5.25 of Ref. 2).*

$-D\,\nabla^2\Phi/\,\Phi = [(k_\infty/k) -1]/L^2 = B_g^2$ *(from Eq;s, 6.8 and 6.10 of Ref. 2)*

The left side is the material buckling, B_m, and the right side is the geometric buckling, B_g. When the two are the same, the configuration is critical.

By comparing the material and the geometric buckling we can determine the dimensions at which the configuration would be critical (when $B_m = B_g$). Conversely if the dimensions are significantly different, the configuration can be super or subcritical, with how much super or subcritical being indicated by the difference between the values.

For this hand calculation, the goal of which is to determine approximately whether the configuration of the assemblies, within the reactor , can be critical, a one group fast reactor approximation is used [2].

The configurations analyzed is shown in the figure at the end of this calculation:

1. A single assembly, modeled as a bare, un-reflected configuration, for which we know the answer is that such configuration is not critical.

2. A multiple assembly configuration, in which the assemblies are separated from one another by a gap of sodium, the dimension of which can be varied to allow for more or less leakage of neutrons. For this analysis, the initial configuration chosen has a minimum gap equal to 4 cm. This minimum gap is created by the tubes which penetrate the redan bottom and extend the hot pool into the cold pool, and into which the spent fuel assemblies are inserted. Each assembly has the widest dimension equal to 20.5 cm (corresponding to a flat-to-flat dimension of the hexagonal duct of 17.75 cm). Each assembly would fit in and be supported by a circular tube the internal diameter of which is assumed to be 22 cm , allowing a possible distortion of the assembly. This is shown in Figure A.1, derived from scaling the tube diameter

and separation from Figure A.2. Therefore, there is uncertainty in the diameters and separation, but the values chosen represent the shortest separation and should therefore be conservative, from the criticality standpoint . Each circle representing the thimbles is separated from the adjacent circle by 3.5 cm and the thickness of each circle is assumed to be 0.25 cm, so the closest distance possible between adjacent assemblies is 4.0 cm.

Each fuel assembly at discharge, 10 and 50 years later has the following quantities of fissionable and fissile material, and fission products in grams/assembly . In addition, each assembly contains what was originally 10% by weight of Zr in the original loading of U-10Zr [1]:

Table 1. Major Isotopes in Spent Fuel Assembly (Grams/assembly from Reference 1)

Isotope	At Discharge	10Years	50 year	Comments
U234	9.97E+00	1.32E+01	2.38E+01	
U235	1.647E+04	1.65E+04	1.65E+04	Fissionable
U236	2.81E+03	2.81E+03	2.81E+03	
U237	7.07E-01	5.8E-07	7.32E-08	
U238	1.945E+05	1.94E+05	1.94E+05	Fissionable and Fissile
Np237	2.47E+02	2.48E+02	2.5E+02	
Pu238	4.30E+01	3.99E+01	2.9E+01	
Pu239	1.041E+04	1.05E+04	1.05E+04	Fissionable
Pu240	6.66E+02	6.66E+02	6.62E+02	
Pu241	2.61E+01	1.62E+01	2.36E+00	
Pu242	1.25E+00	1.25E+00	1.26E+00	
Am 241	5.25E+00	1.51E+01	2.74E+01	
Other actinides	0.839	0.415	0.857	
Fission Products	1.97E+04	1.97E+04	1.97E+04	

<u>1. Single “bare” assembly</u>

The weight of Zirconium in each assembly is 24,238,000*0.1/99=24,483 g. In addition, in the fuel region of the core, on per assembly basis, there is HT9 material present in the cladding , the hexagonal duct and the spacing wire. There are 217 pins in each assembly and the diameter of the pin is 1.041cm, with a thickness of 0.5 mm. Therefore, the total volume of HT9 in the cladding equals 217xπ/4 $(1.041^2-0.941^2)$* 150 cm^3=5066 cm^3. With the HT9 density being 7.87 g/cc, this equates to a weight of 3.99E+4g. The duct work is 0.3 cm thick, so it volume over the length of the fuel region equals 6 *10.25*150*0.3=2767.5 cm^3 and its weight is 2.178E+04 g . The weight of the spacing wire is approximately 3.0E3g. the weight of the fuel at discharge is 2.445E5 g. The total weight of the fuel section of the assembly is 3.357E+05 g. However, the assembly is assumed to be in a sodium medium, therefore the space between the pins is full of sodium, and the weight of the sodium is calculated to be equal to the internal volume of the interior of the ductwork (38,209 cc) less the volume occupied by the pins 27,704cc and spacing wire 341 cc), or 10,160 cc. Assuming conservatively a low density for sodium (0.860 g/cc) the weight of sodium in the assembly is 8.74kg . The total weight of the assembly fuel region is 3.443E+05 g

In each assembly the distribution of the fissionable and fissile material and the rest of the isotopes and structural material is as indicated in Table 2

Table 2 Material distribution and One Group Reactor Physics Cross Sections [2]

	% in Ass.bly	Atom Density	σ_f (barns)	Σ_f cm^{-1}	σ_a	Σ_a	σ_{tr}	Σ_{tr}	ν	η
U235	4.9	2.01E21	1.4	0.00282	1.65	0.00332	6.8	0.0139	2.6	
U238	57.8	2.34E22	0.095	0.00222	0.255	0.00595	6.9	0.161	2.6	1.68
Pu239	3.13	1.25E21	1.85	0.00232	2.11	0.00264	6.8	0.00851	2.98	
Fe	18.3	1.61E22	0	0	0.006	0.0000964	2.7	0.0434		
Zirconium	7.28	3.10E21			0.015	0.000047	3.2	0.010		
Sodium	4.34	9.77E20			0.0008	7.82E-6	3.3	0.00323		
MA	1.2	See Table 3								
Others (FPs)	5.75	See Table 3								

To determine the approximate values for the "others", absorption and transport cross section of isotopes representing groups of the other "not listed" actinides and the fission products have been obtained from reference 4, by also comparing the values given above for one group to the corresponding values for two groups, where group 1 is for neutron energies above 1.35 MeV, and group 2 is for energies below 1.35 Mev , as shown in Table 3; and also examining the multigroup cross section given in tables 7-1 through 7.3 of Ref. 4. For the fission products, examination of the quantities in the fuel shows that the dominant isotopes are Mo (1.5kg), Sr (3.2 kg), Ru (1.25kg), Zr (2.1kg), Cs-Ba (2.8kg),and the Lanthanides (At. Weight 143-150, 4.3 kg). So as representative of the FPs, Zr is chosen, with cross sections derived from examining the multigroup cross section of the same tables.

The atomic density of each of the above is determined from: N = Percentage in second column of Tables 2 or 3, times density of element (i.e., 16.02g/cc for the fuel [5], 7.87 g/cc for Fe, and 0.86 g/cc for Na) times Avogadro's number divided by the atomic weight

η is calculated from$[\nu\,\Sigma_{f(235)}+\nu\,\Sigma_{f(238)}+\nu\,\Sigma_{f(239)}]/[\Sigma_{a(235)}+\Sigma_{a(238)}+\Sigma_{a(239)}]=1.68$

The geometric buckling for a cylinder of radius R and height H is given by the following equation:
$B_g^2 = (2.405/R)^2 + (\pi/H)^2$ *(From Table 6.2 of Ref. 2)*

The material buckling for one group of a homogeneous configuration $k_\infty = f * \eta$, where f is the fuel utilization factor defined as the total absorption in the fuel divided by the total absorption.
f = 0.976
$k_\infty = 1.64$

Table 3 Comparison of One Group *(Table 6.1 Ref. 2)* and Two Groups Reactor Physics Constants *(Table 7-5 of Ref. 4)*

	% in Ass.bly	Atom Density, N	σ_f (barns)	Σ_f cm^{-1}	σ_a	Σ_a	σ_{tr}	Σ_{tr}	ν	η
U235	4.9	2.01E21	1.4	0.00299	1.65	0.00332	6.8	0.01454	2.6	
			1.29		*1.58*		*4.5*		*2.7*	
			1.44		*0.28*		*7.2*		*2.5*	1.68
U238	57.8	2.34E22	0.095	0.002372	0.255	0.00595	6.9	0.1723	2.6	
			0.524		*0.36*		*4.6*		*2.6*	
			0.005		*0.19*		*7.1*		*2.47*	

Pu239	3.13	1.25E21	1.85	0.002487	2.11	0.00264	6.8	0.00851	2.98	
			1.95		*1.0*		*4.6*		*3.1*	
			1.78		*0.97*		*7.0*		*2.93*	
Fe	18.3	1.61E22	0		0.006	0.0000964	2.7	0.0434		
			0		*0.005*		*2.0*			
			0		*0.006*		*2.8*			
Other actinides (Thorium)	1.13	4.99E20	0	0	0.25	0.000117	7.8	0.00367		
Zirconium	7.28	3.1E21			0.015	0.000047	3.2	0.010		
FP(Zirconium)	3.15	1.36E21			0.015	0.0000203	3.2	0.00434		
Sodium	4.34	9.77E20	0		0.0008	0.00000762	3.3	0.002323		

Two groups of constants are shown in Italics. In parenthesis the isotopes representing the other actinides and the Fission Products are shown

L^2 for a bare configuration = D / Σ_a and $D = 1/ (3\Sigma_{tr)} = 1/(3*0.248)=1.34$ therefore $L^2 = 1.3123/0.0115 = 110.07$

For a critical configuration where k =1, $B_m^2 = (k_\infty - 1)/ L^2$ or 0.64/110.07 = 0.0058.

The equivalent radius of the hexagon representing the fuel is one for which the area equals that of the hexagon, and thus the R= 9.316 cm. With R=9.316 and H=150, the geometric buckling $B_g^2 = 0.067$.

The geometric buckling is already significantly greater than the material buckling, and the configuration is not critical. In order to be critical, the configuration would have had to have an effective radius of :
$(2.405/R)^2 = 0.0058 – 0.000438 = 0.0054$ so R = 2.405/0.0735=32.73 cm

2. Assemblies spaced a minimum of 4 cm apart in a sodium medium

If the assemblies were pressed together, for instance assume 4 assemblies), the masses of isotopes are increased, in this example they quadruple, but the atom densities and the various cross sections remain nearly the same. Therefore, the material buckling does not change. However, the geometric buckling now has a greater effective radius (equivalent to a cylinder having the same cross-sectional area as four hexagons). Each hexagon has a cross section area of 272 cm^2, so the four together have an area of 1,088 cm^2. The effective radius corresponding to that area is 18.61cm.

The geometric buckling is 0.0172, which is still greater than the material buckling, and this configuration should not go critical. The number of assemblies required for a configuration to be critical with the assemblies directly adjacent to one another is about 11 assemblies. With 12 assemblies the geometric buckling would have a value of 0.004986, which is less than the material buckling value of 0.00616. So, a 12 assembly could be supercritical. Of course, we plan to store a much larger number, but with space between them filled with sodium.

Anyway, in the case of the in-vessel storage, the assemblies are always at a minimum distance of 4 cm from one another, so there is a significant amount of sodium (and some steel -but not much of the latter) between them. This situation is modeled in this calculation by two configurations: (1)as a square lattice having four assemblies at each corner of the lattice, with a cross between them each of the arms of the cross being 4 cm thick (shown as Figure 2B), and (2) the same configuration surrounded by sodium having a thickness of 2 cm (to simulate the effect that the assemblies also have leakage beyond that provided by the internal cross) as shown in Figure 2C.

In both cases, the major difference from Tables 2 and 3 are the atomic densities of the fissile isotopes, other isotopes, isotopes, HT9, zirconium and sodium. In case (1) an additional amount of sodium equal to 3.616e+04 g is now present, and in case (2) the additional amount is 5.63E+04g.

Table 4 A Case (1) Storage Configurations Constants

	% in Conf.	Atom Density	σ_f (barns)	Σ_f cm^{-1}	σ_a	Σ_a	σ_{tr}	Σ_{tr}	ν	η
U235	4.42	1.81E21	1.4	0.00254	1.65	0.00299	6.8	0.0127	2.6	1.636
			1.29		*1.58*		*4,5*		*2.7*	
			1.44		*0.28*		*7.2*		*2.5*	
U238	52.2	2.12E22	0.095	0.00204	0.255	0.00547	6.9	0.143	2.6	
			0.524		*0.36*		*4.6*		*2.6*	
			0.005		*0.19*		*7.1*		*2.47*	
Pu239	2.8	1.13E21	1.85	0.00209	2.11	0.00239	6.8	0.00769	2.98	
			1.95		*1.0*		*4.6*		*3.1*	
			1.78		*0.97*		*7.0*		*2.93*	
Fe	16.5	1.45E22	0		0.006	0.0000869	2.7	0.0412		
			0		*0.005*		*2.0*			
			0		*0.006*		*2.8*			
Other actinides (Thorium)	1.02	4.7E20	0		0.25	0.000117	7.8	0.00367		
Zirconium	6.58	2.83E21			0.015	0.0000425	3.2	0.00906		
FP (Zirconium)	2.85	1.02E21			0.015	0.0000184	3.2	0.00392		
Sodium	11.8	2.66E21			0.0008	0.0000213	3.3	0.00877		

Two groups of constants are shown in Italics. In parenthesis the isotopes representing the other actinides and the Fission Products are shown

f= 0.973

η is calculated from[$\nu\ \Sigma_{f(235)+}\ \nu\ \Sigma_{f(238)}\ +\nu\ \Sigma_{f(239)}$]/ [$\Sigma_{a(235)}\ +\Sigma_{a(238)+}\ \Sigma_{a(239)}$]=1.67

k_∞ = 0.973*1.67=1.63

$D = 1/(3\ \Sigma_{tr)} = 1.45$

$L^2 = D / \Sigma a$ =1.43/0.011=130.25

$B_m^2 = (k_\infty - 1)/ L^2$ or 0.63/130.25= 0.00485

The equivalent radius of the configuration is the one corresponding to the overall cross sectional area of the configuration, as shown in Fig . A.1B it is 20.9 cm.

With this radius and a height of 150 cm, the geometric buckling is $(2.405/20.9)^2 + (\pi/150)^2 = 0.01368$. Since the geometric buckling is still greater than the material buckling, this configuration could not go critical. Moreover, first the 4 cm distance is the minimum between any assembly, but in reality, it is mostly greater by at least another 1-2 cm, and in order to potentially be able to go critical it would require having an equivalent radius determined from :

$(2.405/R)^2$ = 0.00485-0.0004386 = 0.00441 R= 2.405/0.06642= 36.21cm, whereas ours is only 20.9 cm.

Nevertheless, it is good to check how far we may be from criticality by looking at the second configuration, in which the equivalent radius is 23.15 cm.

Table 4 B Case (2) Storage Configurations Constants

	% in Conf.	Atom Density	σ_f (barns)	Σ_f cm$^{.1}$	σ_a	Σ_a	σ_{tr}	Σ_{tr}	ν	η
U235	4.3	1.877E21	1.4	0.00247	1.65	0.00291	6.8	0.0122	2.6	
			1.29		*1.58*		*4,5*		*2.7*	
			1.44		*0.28*		*7.2*		*2.5*	
U238	48.62	1.87E22	0.095	0.00178	0.255	0.00478	6.9	0.129	2.6	1.678
			0.524		*0.36*		*4.6*		*2.6*	
			0.005		*0.19*		*7.1*		*2.47*	
Pu239	2.7	1.09E21	1.85	0.00202	2.11	0.0023	6.8	0.00741	2.98	
			1.95		*1.0*		*4.6*		*3.1*	
			1.78		*0.97*		*7.0*		*2.93*	
Fe	16.5	1.45E22	0		0.006	0.0000869	2.7	0.0391		
			0		*0.005*		*2.0*			
			0		*0.006*		*2.8*			
Other actinides (Thorium)	1.05	4.36E20	0		0.25	0.0001095	7.8	0.0034		
FP (Zirconium)	4.46	1.92E21			0.015	0.0000288	3.2	0.00614		
Sodium	16.2	3.65E21			0.0008	0.0000292	3.3	0.012		
Zirconium	6.17	2.65E21			0.015	0.0000398	3.2	0.00849		

Two groups of constants are shown in Italics. In parenthesis the isotopes representing the other actinides and the Fission Products are shown

In this case f = 0.975, η=1.67, and k_∞ =1.63; $D = 1/(3\ \Sigma_{tr})$=1.50 and L^2 = 141.9. Therefore B^2_m =0.0044

B^2_g = 0.01123. The difference between the two is now a bit less than before and this configuration is as or less subcritical than the first one.

In this configuration , with Keff= $(\gamma\ \Sigma_f/\ \Sigma_a)/(1+L^2B_g^2)$ =1.72/(1+141.7*0.01123)=0.63

<u>Conclusion</u>

The present proposed configuration , with a minimum spacing between the fuel assemblies of 4.0 cm, but most of the time considerably larger than that, should not be able to go critical. In fact, it is possible to reduce the spacing and make the storage facility more compact. The calculations above indicate that a spacing of approximately 2 cm should more than suffice. Nevertheless, this is a one group calculation, with a number of assumptions, judged to be conservative in regard to the absorption cross section of the fission products, but ignoring the fission cross sections of some of the actinides which can also be fissioned by fast neutrons (the reason being that these are present in much smaller quantities than those explicitly considered). The geometric assumptions made in the calculation are judged to be conservative and to more than compensate for the lack of detailed considerations of every isotope in the fuel. The results indicate that there should be no concern with criticality in the proposed configuration. It is my recommendation, however, that a detailed criticality analysis be done, with multigroup cross sections and an appropriate code, to conclusively verify this result, before finalizing the configuration of the fuel storage in the reactor vessel

REFERENCES FOR ANNEX 1

[1] T.K. Kim "Decay Heat of 286 MWW ARC Core Fuel Assembly (Rev 00)", Nuclear Science and Engineering Division, Argonne National Laboratory, August 23, 2021

[2] J. R. Lamarsh, "Introduction to Nuclear Engineering", Addison-Wesley Pub Co. , 1975

[3] R. C. Iotti "Calculations of decay heat, dose rates, and criticality concerns with ARC 100 spent fuel disposition" August 21, 2021

[4] US Atomic Energy Commission report ANL -5800, 2nd Ed., 1963

[5] D. Porter," Density of U-10 w/o Zr Materials , Rev 2",INL/EXT-17-41917, Rev 2, April 20017 .

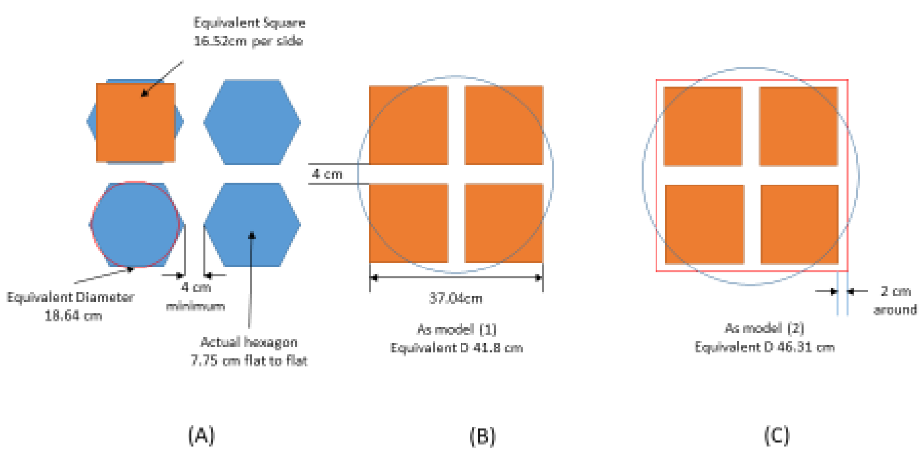

Figure A.1 Proposed configuration of the spent fuel storage showing spaces for spent fuel assemblies

[[Figure A. 2 Possible layout of spent fuel in reactor vessel]]EXPC