

Spent Nuclear Fuel Storage in a Molten Salt Pool

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Abstract

In the aftermath of the tsunami in Japan and the disastrous events that followed at the Fukushima Daiichi generating station, the industry's attention has largely been turned to the spent fuel pool. While the nuclear core is sealed from the outside world by the steel reactor vessel and a containment building several feet thick of steel-lined, reinforced concrete, spent nuclear fuel sits in a pool that, while heavily reinforced for structural purposes, is largely unprotected. It can be exposed to the outside through events such as the hydrogen explosions that occurred at Fukushima. Increasing the robustness of the spent fuel pool (SFP) buildings as well as spent fuel cooling in the event of offsite power loss has become the focus of much discussion. This exploration focuses on the latter.

Herein an approach to fuel pool cooling is explored that has the potential to extend the length of time a SFP could maintain fuel integrity without offsite power. It is suggested that spent fuel pools be constructed to hold a molten salt, rather than water. During normal operations, the molten salt would be circulated through a heat exchanger to remove decay heat just as is done in light water reactor SFPs with water. The molten salt has several benefits over water, including being a poor neutron moderator eliminating criticality concerns. The most important factor is that molten salt surrounding spent fuel can absorb heat through conduction without boiling and uncovering the fuel. The boiling points of available salts are well above the melting points of various fuel assembly materials.

COMSOLTM,^A multi-physics heat transfer models were built to predict the transient thermal behavior of spent fuel surrounded by a particular salt, LF-KF-NaF, also known as FLiNaK. These models suggest that even with no heat removal to an ultimate sink, a spent fuel pool with a total heat load of 3MW decay heat will not reach melting temperatures for 10 days, in which time water would have long since boiled away leaving the assembly with no means but radiation to stay cool. If the heat load can be reduced to 2MW, the models suggest the fuel will be safe over 15 days after a station blackout. This figure does not consider additional heat removal that would be achievable using an emergency cooling system driven by natural circulation.

^A COMSOL, Inc., Burlington, MA

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Purpose and Background

On March 11, 2011 the latest nuclear disaster occurred at the nuclear power plant of Fukushima Daiichi in Japan. A 9.0 magnitude earthquake was measured 112 miles off the coast.¹ All of the operating units (1, 2, and 3) responded by automatically scrambling but the earthquake caused the loss of all off-site power. The diesel generators automatically started and provided emergency power. However, 41 minutes after the earthquake, seven tsunamis arrived at the site. The design basis tsunami height was only set at 5.7 meters but the tsunami that hit the plant measured up to 14-15 meters high. All AC power was lost to units 1-4 but one emergency diesel for Unit 6 provided enough power for Units 5 and 6.¹

The only units operating were Units 1, 2, and 3. Unit 4 had moved all the fuel into the spent fuel pool. Since there was a loss of all power, the water temperature in the reactors of Units 1, 2, and 3 raised high enough to oxidize the Zircaloy which produced high amounts of hydrogen. This hydrogen content continued to build up until there was an explosion in the Secondary containment of Units 1, 2, and 3.¹ Unit 1 suffered damage with the roof collapsing on the operating floor. Unit 2 suffered the least amount of damage with the roof remaining intact. The roof and structure were completely destroyed in Unit 3 and debris had fallen in the spent fuel pool. Although the reactor in Unit 4 was not operating, it still suffered a hydrogen explosion leading to damages done to the roof and structure along with debris falling in the spent fuel pool. The most credible theory for the explosion at Unit 4 is that some of hydrogen left standing in Unit 3 had transferred into Unit 4 through its shared vent stack and piping.¹ The figure below shows the damage done to Unit 4 after the tsunami and earthquake.



Figure 1: Fukushima Unit 4 Spent Fuel Pool Before and After¹

The spent fuel pools in these units had different heat loads which led to different times to when the fuel became uncovered. If the spent fuel ever became uncovered, there could be tremendous amounts of radiation released to the environment. Therefore, it was paramount to keep the fuel covered with water. At normal conditions, the water was circulated in the spent fuel pool to remove heat but with the loss of all power, the water remained stationary. This caused the water to reach very high temperatures which was high enough to boil the water away. There was also fear that structural damage from the earthquake and explosion caused a leak in the spent fuel pool. However, this was proven not to be the case. As mentioned before, Unit 4 had moved all of its fuel into the spent fuel pool. The fuel from Unit 4 was considered fresh fuel which caused further complications because of the high heat load (2.3 MW). It was calculated that the time to uncover 50% of the fuel was only 11 days.¹

In the wake of the unfortunate events in Japan, eyes have turned to increasing the robustness of nuclear power stations, and in particular fortifying and protecting the spent fuel pools. “Beyond Design Basis” events are now challenges that must be resolved in order for people to feel safe trusting the nuclear industry with their safety and future. Whatever the cause of it, an extended loss of off-site power, or total station blackout, is one such Beyond Design Basis event that is being addressed. The design put forth and modeled in this report proposes an innovative solution to the problem of a SFP that has lost its active cooling system. A Fukushima-type event resulting in a total blackout and loss of cooling capabilities in a plant built after the design

suggested herein could have up to 20 days of time before a fuel melt would occur. The design is certainly not without challenges, but our analysis suggests it is worth taking a serious look at.

The scope of this report is not intended to provide quality documentation of a detailed design. The purpose of this report is to demonstrate the validity of the concept as it may be put against the concerns of heat transfer and broad-view logistics. Readers are directed to the section on future work for a summary of considerations left unexplored and an address of likely concerns that the readers may have.

Design Details

Overview

The suggested spent fuel pool(SFP) would be positioned as deep underground as is practical and beneficial, pending a thorough analysis of the accident scenario natural circulation. More on this is discussed later in this section. Figure 2, below, shows the proposed spent fuel pool(SFP) design modification from cut-away side view. Rather than a pool of circulating water, the fuel would be submerged in a pool of molten salt. The design modeled later used LiF-NaF-KF (a.k.a. FLiNaK), because it provided a high thermal conductivity and minimal corrosivity concerns. However, other salts are available and may prove more beneficial reasons discussed later.

The molten salt would be circulated through the assemblies as is done currently in water-filled SFPs. Figure 2 shows the circulation path as the salt (in yellow) is sent to a heat exchanger for cooling. Note all components of the salt cooling loop would be specially designed to accommodate transference of the molten salt as it pertains to corrosivity concerns, heat stress, and accident scenario solidification. The latter concern is addressed in a few paragraphs.

The heat exchanger is cooled by waters from the nearby water body or cooling towers. As may be required for liscencing purposes and equipment purposes, filters, demineralizers, contamination monitors, and other treatment/monitoring components would be placed appropriately as the cooling water enters and leaves the system to the environment.

The reactor vessel is shown in the upper left hand corner of Figure 2. When fuel is unloaded it is transferred to a vertical chute which sends the assembly one at a time to an intermediary chamber where the water is drained and replaced with the molten salt. A section on the fuel transfer process discusses this as well as reloading in detail.

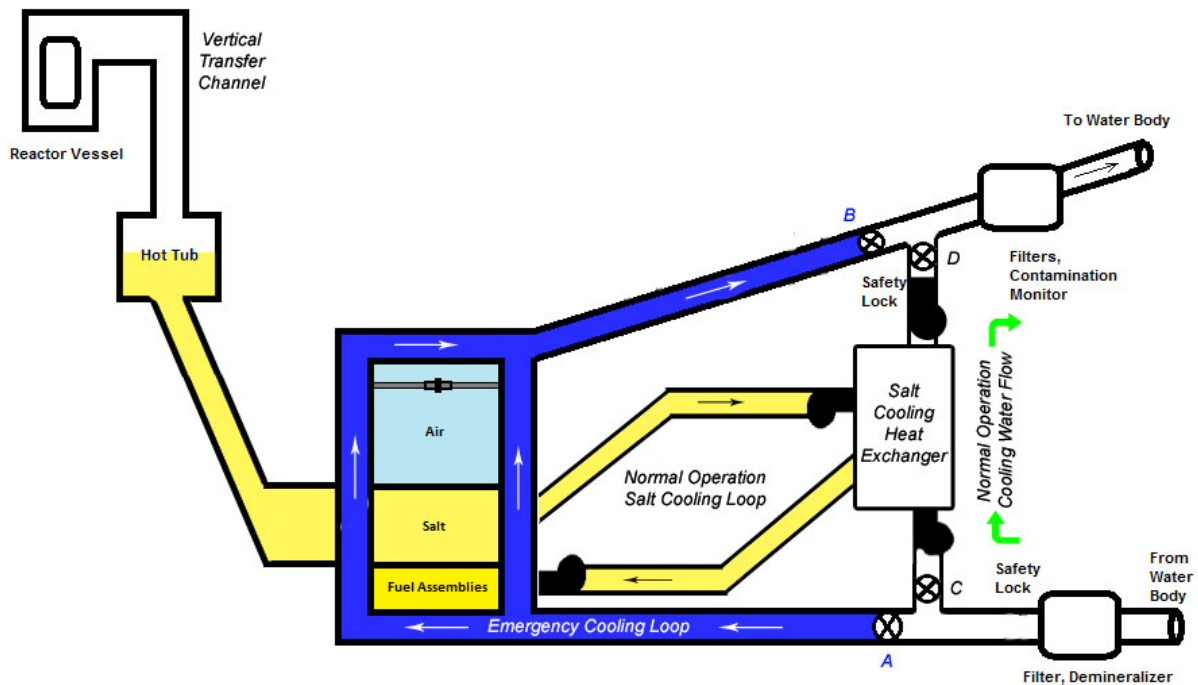


Figure 2: Cutaway side-view schematic of spent fuel pool and related systems. Valves A & B are closed, while valves C & D are open during normal operations.

The emergency cooling water channels, shown in blue, are filled with an inert gas during normal operations. The channels branch off of a point upstream of the cold entrance to the salt cooling heat exchanger, though the valve (A) stays closed during normal operations. They run from that point into the plant horizontally or at a slight upward angle to a point under the center of the SFP. The channel splits into four separate channels which are routed out and up the four sides of the SFP outer walls. On the way up, the pass over conduction plates and fins which draw heat away from the inside of the pool. After reaching the top of the room above the pool, the channels merge together and route the water to a point down stream of the cold exit of the heat exchanger. The valve (B) prior to the merge also remains closed during normal operations. The water channel design as it surrounds the SFP is shown in Figure 3, below.

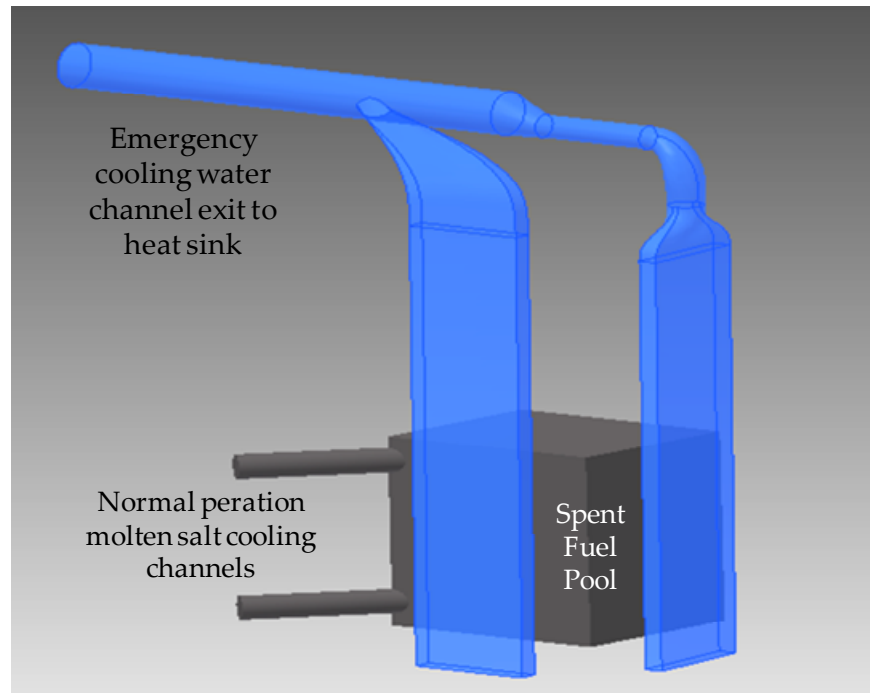


*Figure 3: Emergency cooling water channels surrounding SFP.
Water entrance from heat sink and two vertical channels on opposite side not shown.*

In the event of a loss of off-site power accident, the following sequence is initiated:

- Dedicated battery backups are activated to accomplish valve openings/closings and other operations below.
- Valve A is opened, letting water rush into the emergency cooling channel. Gas is allowed to escape via a vent near Valve B.
- When the water level has reached Valve B, it is opened while valves C & D are shut bypassing the heat exchanger. All pumps are tripped at this time ending the forced circulation and cooling of the molten salt.
- Jet pumps, powered by the battery backups, are activated at optimal points along the water path around the spent fuel pool, increasing the flow rate and heat removal via the conduction plates in the vertical channels shown in Figure 3.
- When the battery backups have expired, the emergency water flow will be unhindered by the pumps and will be driven by the density change resulting from heat transfer to the water.

Removable conduction fins can be designed to be suspended over the fuel racks to provide a path of high thermal conductivity for heat escape. Interlocking cleats on the pool walls would hold the fins in place just above the fuel. These mounting cleats would be fastened to aluminum plates which protrude through the walls into the vertical water channels. The fins would be constructed of steel-reinforced aluminum and clad in Hasteloy-N to protect the Aluminum from corrosion. During normal operations they would be stored outside of the pool. When a loss of off-site power occurs, operators would remotely use the SFP overhead crane to position the fins in place, increasing heat removal through conductivity.

The emergency cooling water channels and removable fins are suggested to provide some decay heat removal and augment the length of time the SFP would remain safe after losing power. However, these measures are not at the heart of the molten salt SFP concept. In fact, as is discussed in the section on *Thermal Analysis*, conduction was shown to be a largely ineffective means of removing decay heat. Some heat removal is better than none, thus the need for a passive cooling system. However, the primary benefit gained from using the salt pool is the heat storage capacity of the salt at temperatures all the way up to the fuel structure melting temperatures.

Molten Salt Selection

When choosing a salt compound for a coolant purpose, it is important to find a salt that will maintain its chemical stability under extreme temperatures, has melting and vaporization points appropriate for the application, and will not cause corrosion or galvanic drift with the alloys in use. All of the above properties and effects were taken into account when considering different candidate salts. Below is a table of the candidate salts.

Table 1: Summary of candidate salt properties(copied without permission)²

Salt ^a	Formula weight (g/mol)	Melting point (°C)	900°C vapor pressure (mm Hg)	Heat-transfer properties at 700°C			
				ρ , density (g/cm ³)	$\rho^{\circ}C_p$, volumetric heat capacity (cal/cm ³ ·°C)	μ , viscosity (cP)	k , thermal conductivity (W/m·K)
LiF-NaF-KF	41.3	454	~ 0.7	2.02	0.91	2.9	0.92
NaF-ZrF ₄	92.7	500	5	3.14	0.88	5.1	0.49
KF-ZrF ₄	103.9	390	1.2	2.80	0.70	< 5.1	0.45
LiF-NaF-ZrF ₄	84.2	436	~ 5	2.92	0.86	6.9	0.53
LiCl-KCl	55.5	355	5.8	1.52	0.435	1.15	0.42
LiCl-RbCl	75.4	313	--	1.88	0.40	1.30	0.36
NaCl-MgCl ₂	73.7	445	< 2.5	1.68	0.44	1.36	0.50
KCl-MgCl ₂	81.4	426	< 2.0	1.66	0.46	1.40	0.40
NaF-NaBF ₄	104.4	385	9500	1.75	0.63	0.90	0.40
KF-KBF ₄	109.0	460	100	1.70	0.53	0.90	0.38
RbF-RbF ₄	151.3	442	< 100	2.21	0.48	0.90	0.28

The first and most important physical properties taken into consideration were the melting and vaporization points. Because salts possess high heat capacity, the temperature drop in a salt is typically smaller than for other high-temperature coolants.² For the reference conditions analyzed, considerable thermal margin to freezing exists. The minimum steady-state temperature in the loop under reference conditions is ~680°C, which represents a margin for 130°C for FLiNaK salt. Grace periods for the onset of freezing were estimated to occur between 65–80 min for an off-site power loss. These times were reasonable for taking actions to prevent the problems that would occur with considerable freezing of the salt.² Therefore, the properties and freezing points associated with FLiNaK (454°C m.p.) may be acceptable for use. Because no single component salt freezes at a sufficiently low temperature, multi-constituent mixtures of salts are required. In general, the lowering of a freezing point occurs with the addition of the first salt to a pure component.² Additional lowering can occur by adding a third component. The heat capacities of all the salts mixtures are available. When possible, heat capacity values were evaluated at 700°C. In some instances, no accurate temperature dependence was available from the experimental database. In general, the variation of heat capacity of molten salts with temperature is small. It is difficult to measure small values of viscosity at high temperatures, and special methods have been developed for this purpose.²

Cost factors must be considered with respect to selection of a molten salt. However, we cannot predict this cost for all salts, because some of the constituents of candidate salts are not commodity chemicals, and the cost associated with the deployment of significant numbers of heat-transfer loops would bury the markets and would change the current price. It is also possible that the market could change the specialty prices associated with RbF and KF compounds. Rubidium has an unusual position with respect to markets. While the world market for rubidium is extremely small. It ranks as the 23rd most abundant element on earth. Rubidium is more abundant than copper, lead, and zinc.²

The corrosion data for chloride and fluoroborate salts is much more limited than the fluoride data, especially for temperatures above 650°C. For example, the effect of chromium content in the alloy does not seem to be an important factor, and the effect of temperature is not understood.² Unexpected variability in the tests very likely reflects variability in the purity of the starting materials and the degree to which impurities were excluded from the loop during operation. The corrosion rates are rather high (2.1-10.7 mil/yr) for these salts at a relatively low temperature (~550°C). These rates are similar to those experienced with fluoride salts in contact with stainless steels and Inconel at ~800°C and are much higher than those experienced with Hastelloy-N (0.55-1.2 mil/yr) in contact with fluoride salts at temperatures as high as 815°C.² After a careful analysis of a multitude of candidate salts LiF-NaF-KF (FLiNaK) was chosen as the material to be used in the spent fuel pool. The values given in the table have been found to be fairly accurate over wide temperature ranges. Overall, because FLiNaK is clearly superior in heat transfer to all other salts, therefore, there is little reason to consider other moderately expensive salts. It was for this reason that FLiNaK was chosen for use in this model.²

FLiNaK is the name of the ternary eutectic alkaline metal fluoride salt mixture LiF-NaF-KF (46.5-11.5-42 mol %). It has a melting point of 454 °C and a boiling point of 1570 °C. FLiNaK salt was researched heavily during the late 1950s by Oak Ridge National Laboratory as potential candidate for a coolant in the molten salt reactor because of its low melting point, its high heat capacity, and its chemical stability at high temperatures.²

FLiNaK is a fluoride salt, and like all salts, causes corrosion in most metals and alloys. There are three mechanisms through which they can cause corrosion: impurities in the salt, temperature

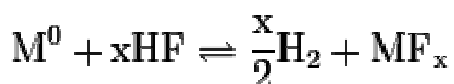
gradients in the salt system, and chemical activity gradients. The main oxides that can cause corrosion in these systems are O_2 , H_2O . Corrosion is caused through the following reactions:



Water and fluoride ions create acid and oxygen.



Water and fluoride ions create acid and hydroxide.



Acid oxidizes M, where M is any suitable metal (ex: Cr, Fe, Ni).²

All of these reactions work together to create hydrofluoric acid, an acid that is highly corrosive to many metals. All of these reactions can be reversed by adding a combination of hydrogen gas to the molten salt, yielding water, which can be boiled off. Theoretically, if contaminants could be taken out of the salt, no corrosion would occur. This process is accelerated by the high temperatures encountered in a nuclear reactor and is one of the main problems in a molten salt reactor. Some alloys, such as the nickel-based Hastelloy-N were engineered specifically for the molten salt reactor experiment and proved highly resistant to corrosion by molten fluoride salts. For this reason, Hastelloy-N was chosen to line the ‘hot tub’ because of its resistance to corrosion from HF.

Transfer To/From Reactor Vessel

Transferring the fuel assemblies to and from the pool and the reactor will be very similar to the transfer method used in a standard spent fuel pool. The challenge lies in keeping the water from the reactor coolant loops completely isolated from the FLiNaK salt in the pool. To accomplish this there will be an intermediate chamber that will be used to clean the fuel rods between the SFP and reactor, known as the Hot Tub shown as the light blue chamber in Figure 4.

The chamber itself is designed specifically so that it can only hold one fuel bundle at a time 4 meters in length at a time. It is 5 meters in length and has a radius of 0.54 meters. The overall volume of the chamber is approximately ~4.5 cubic meters (~1200 gallons) when completely filled.

The Hot Tub's primary purpose is not only to clean the fuel rods while in transit, but it is also designed to minimize any contact of the salt with water. When FLiNaK salt comes into contact with water it produces a by-product of hydrofluoric acid, which is very corrosive and ultimately catastrophic if left in contact with the Zircaloy fuel rods.² This is the case when the pH of the hydrofluoric acid approaches 3 or less.² Thus, the inside of the Hot tub is lined with Hastelloy to minimize corrosion from the possible formation hydrofluoric acid.²

Another important duty of the Hot Tub is that it keeps the fuel rods at a safe temperature while they are being cleaned and also to shield the fuel bundle in the absence of water. At Fukushima, the cladding integrity of the fuel bundles in the spent fuel pool remained uncompromised for until around two hours.² The bundle in the Hot tub will be completely uncovered for only a matter of minutes at most as it is cleaned. The Hot tub will be shielded with an adequate layer of concrete to prevent radiation into the environment.

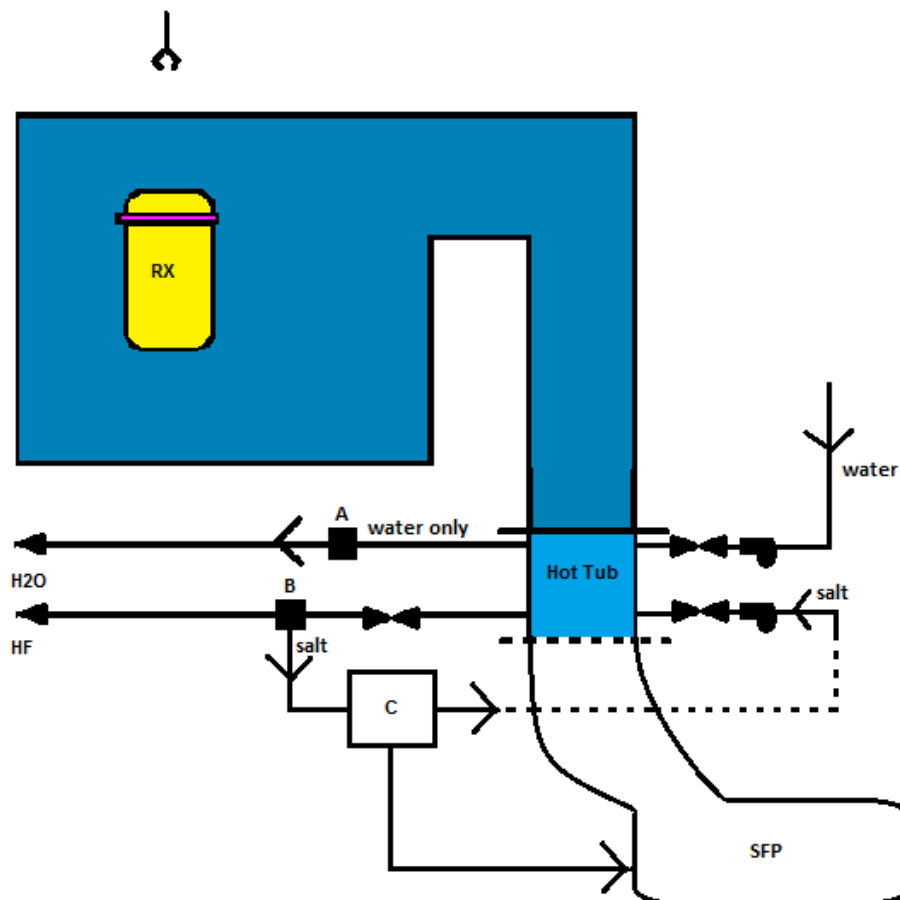


Figure 4: Schematic of the transfer chamber's operation loops (Side view cut out)

The core unloading process begins similarly to the standard procedure used in spent fuel pools today. Containment is flooded and each fuel bundle is taken out of the core by a crane suspended above the reactor vessel.² The crane then moves the fuel bundle to an elevator where it is lowered to the Hot Tub's top containment door. The Hot Tub will be flooded with water, and the elevator will lower the bundle and place it securely onto a rack in the Hot Tub.

Next, the top containment door will close, and initially the fuel bundle will sit in its rack. Then the Hot Tub will be evacuated of all water through six evacuation tubes (Figure 5) spaced evenly, or 0.83 meters apart, along the length of the Hot Tub. All water must be evacuated before the salt is allowed into the Hot Tub out of "Line A" in Figure 4, and any residual water vapor or droplets can be ignored. The fuel bundle should only be uncovered for a few minutes, and the cladding will not be compromised in this short amount of time.

After allowing all of the water to drain from the chamber, the valves close on the water nozzles, the salt valves open, and the molten FLiNaK is allowed in through six different nozzles along the same side that the water entered through (Figure 5). It will fill the container until the fuel bundle is completely covered again, but this time with pure, molten salt.

The water that has been drained is eventually filtered, treated, and it is ultimately discarded from the plant. Opposite the water inlet nozzle side of the Hot Tub is the water exit nozzle side which is set up much the same way as the feed side. There will be an additional set of six valves on each side of the chamber for the molten salt. Total there will be two sets of six exit and inlet nozzles spaced roughly .83 meters apart, or twelve nozzles on each side of the Hot Tub for a total of twenty-four nozzles (Figure 5). One set of six will be opened for water evacuation and on the opposite side of the chamber is another set of six for draining water from the chamber. The other set of nozzles will be used to evacuate and filter the salt solution composed not only of salt, but of water and any negligible amounts of hydrofluoric acid that may have formed show as Line B in Figure 2.

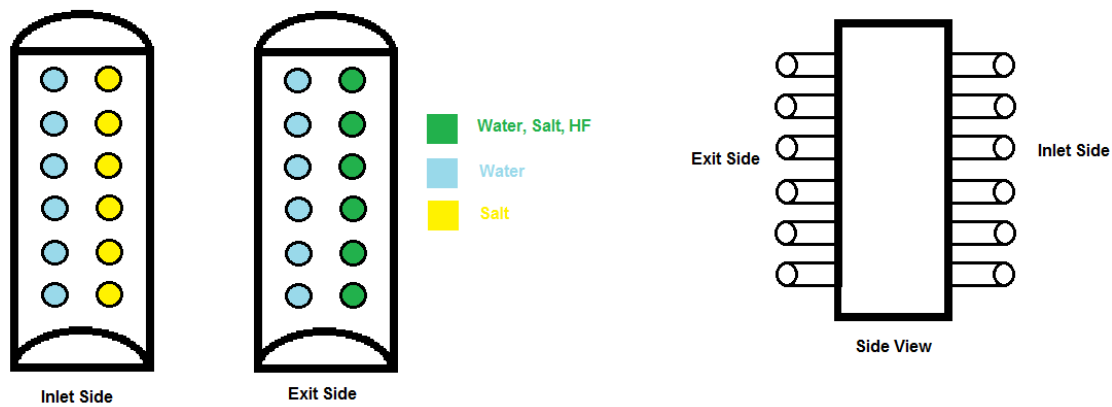


Figure 5: Cut-away view of the hot tub cylinder and side view

The second set of six exit valves lead to the first filter where the FLiNaK salt is separated from any excess water. The salt then heads to an additional intermediate chamber (Filter C Figure 4) where it will be further filtered and made ready to be recycled into the spent fuel pool. The water/hydrofluoric acid mixture continues on to a separate (Line/Filter C Figure 4) where the hydrofluoric acid is separated from the water and discarded as waste, or sold as it is a highly valuable industrial chemical. The water will be treated and discarded from the plant.

Once the fuel bundle has been completely covered with FLiNaK salt by filling the entire Hot Tub with salt, it is ready to be admitted to the spent fuel pool. The bottom containment door is opened, and the automated up-ender carries the fuel assembly to its predetermined location where it will sit until either ultimately discarded as long term waste, or placed back into the reactor.

Reloading the fuel assemblies will be principally the same process as unloading from the reactor. The process of draining and adding salt and water will be reversed, but it will operate much the same way. Since the salt is denser than water, and there is likely to be residual salt in between the rods, the Hot Tub will undergo “rinse cycles”. The rinse cycles will consist of draining salt and adding water multiple times to ensure that the fuel bundle is adequately cleaned to be returned to the core.

Some issues with this design are shielding the fuel from the environment in the absence of water, minimizing the formation of hydrofluoric acid, and keeping it cool while uncovered. Purposely uncovering nuclear fuel seems very counter-intuitive and it is generally not ideal, but the chamber will be more than adequately shielded on the outside of the chamber. Other than that, the Hot Tub will be located sub-surface and outer shielding will be surrounded by soil, which should provide an excess of radiation shielding.

The formation of hydrofluoric acid is the most important issue because hydrofluoric acid is one of the most corrosive acids, even to metals such as Zircaloy and stainless steel. The inside of the chamber will be lined with Hastelloy which is corrosion-resistant to hydrofluoric acid.² Initially, the idea was to simply mix water and the FLiNaK salt in the chamber at the same time until the salt or the water was pure enough for transfer, depending on whether the fuel rods were unloading or loading. This would make sure that the fuel would remain covered, but at the expense of a large production of hydrofluoric acid. This ultimately doesn't work well unless a buffer is added or another solution to make it more basic. This adds another step of filtration that would not only add unnecessary complications, but also increase costs.

The fuel assembly cannot be uncovered for longer than a few minutes to prevent the cladding from becoming too hot. The time that it is uncovered is minimal, and there should be no problems with the Zircaloy-4 melting or having any catastrophic failures.

The decision was made to completely evacuate the chamber and expose the fuel temporarily to minimize hydrofluoric acid exposure and subsequent corrosion. The risks of melting the cladding or having inadequate shielding are negligible, especially when compared to the corrosive capabilities of hydrofluoric acid. The calculations made in Table 2, on the next page, shows that even when the fuel assembly is left completely uncovered, the cladding does not become close to the melting point of Zircaloy-4 until after ten minutes. In the Hot Tub, the fuel will not be uncovered for more than five minutes at the maximum.

Table 2: *Uncovered Fuel Pin Heat-up Calculations. Fuel pin assumed to be at 1073K.*^{1, 3}

Cladding Volume	6.87E-05	m ³
Zirc-4 density at 1073K	6.51E+03	kg/m ³
Zirc-4 mass	4.48E-01	kg
ΔT (2123 – 1073)	1050	K
C_p	354.7	J/kg*K
Q (heat required to melt)	166777.6	J
Heat load per assembly	71000	W
Heat load per pin	268.94	W
Time to melt	620.13	s

Criticality Analysis

Criticality calculations were performed using MCNP version 5, a general Monte Carlo N-Particle Transport Code. The input codes, found in Appendix B, represent three scenarios for analysis: a spent fuel pool filled completely with water at 100°C, a pool filled with LiF-NaF-KF at 100°C, and a pool filled with LiF-NaF-KF at 500°C. The model representing a spent fuel pool filled with salt at 500°C is the most probable condition in this case. Table 3, below, shows the inputs that were used to build the MCNP decks.

Table 3: *MCNP Inputs*^{3, 5}

Fuel Assembly Dimensions (cm)	
Assembly pitch	21.505
Fuel rod pitch	1.265
Pellet outer diameter	0.824
Cladding inner diameter	0.824
Cladding outer diameter	0.952
I/T, G/T inner diameter	1.14
I/T, G/T outer diameter	1.22
Zircaloy 4 Composition(%)	
tin	1.40
chromium	0.10
iron	0.20
oxygen	0.12
zirconium	98.18

The fuel assemblies are modeled as a 3D infinite lattice and are assumed to be fresh bundles at 5% enriched U235. These assumptions are made so that we are certain that in the worst case scenario, there isn't a possibility of a criticality accident. While the assemblies actually placed in the pool will have burnup, it is safely assumed that if the fresh bundle submerged in FLiNaK is less reactive than the same bundle submerged in water, then the same relationship will apply to burned assemblies.

The two extremes of the theoretical temperature scale were modeled to demonstrate the apparent independence of reactivity in the salt to temperature. The temperature 500°C was chosen because the melting point of the salt occurs at 454°C, so this temperature would give some leeway in case the pool was heated more than necessary. The chance of the temperature of the pool ever reaching temperature below 100°C is relatively low, so it's chosen as a safe temperature point to analyze so that we are sure there's not a possibility of k_{∞} being higher. The results of the MCNP calculations are provided below, in Table 4.

Table 4: Criticality Analysis Results

Run	Temperature (°C)	Density of Water (g/cm ³)	Density of Salt (g/cm ³)	k_{∞}	Standard Deviation
1	100	N/A	2.386	0.6602	0.00015
2	500	N/A	2.097	0.6621	0.00015
3	100	0.958	N/A	1.48719	0.00028

When the infinite 3D lattice of fuel bundles is submerged in water, k_{∞} is 1.487. The fact that this is above 1 is okay in this scenario because when we run the same scenario with the pool filled with salt the k_{∞} goes down significantly to 0.6602. This demonstrates that in the worst case scenarios for our pool, the k_{∞} will remain below 1, thus criticality safety will not be a problem.

Model Validation

The thermal analysis is guided by the need to maintain safe temperatures for the constituent materials in the SFP. Table 2 lists the materials in the SFP that must be considered, along with relevant thermo-physical properties.

Table 5: Thermophysical Properties of Materials in the SFP^{4,5}

Material	Density (g/cm ³)	Melt Temp (K)	Thermal Capacity (J/g-K)	Thermal Conductivity (W/m-K)	Reference
Alloy-718	8.19	1483	0.435	11.40	4
Zircaloy 4	6.56	2123	0.292	21.50	4
SS(304L)	7.83	1673	0.500	21.40	4
FLiNaK	2.02	727	2.010	0.92	4
UO ₂	10.82	3120	0.312	2.30	6
Aluminum (A92014)	2.80	780	0.880	154.00	4
Water (for comparrrison)	1	0	4.180	0.60	4

The lowest melting temperature in a typical SFP is that of Alloy-718, which is used in the assembly grid straps. This occurs at 1483K.⁴ While the disaster of a fission product release due to clad melting does not occur until 2123K⁴, we shall consider 1483K to be a failure in terms of determining the blackout life of the SFP. This temperature, reached, may not result in a contamination release, but it would certainly constitute a disaster in terms of cleanup and fuel cost. The aluminum alloy suggested for use in the conduction fins actually melts at 783.15K⁵, which significantly reduces the available heat up margin. This will be discussed in the analysis. However, recall that we will conclude that conduction plays a relatively minor role in the mitigation of fuel heat up. A suggestion for future work is a thorough optimization of all parameters including the time gained from adding the fins compared to the tie lost by lowering the temperature limit.

The normal operating condition is not modeled in this study. It is assumed that decay heat will be adequately removed by the molten salt cooling loop and heat exchanger. Furthermore, the natural circulation in the emergency cooling water channels is not modeled. The schematic and piping system conceptualized are designed to maximize the effects of natural circulation, which

will drive the water flow in a station blackout. However details such as exact dimensions, mass flow, and heat removal are not calculated. Rather than consider an assumptive heat removal resulting from this system, we have opted to exclude it from our calculations.

The remaining analysis is in the heat buildup in the SFP proper. COMSOL™ multiphysics simulation software was used to model a three dimensional SFP. The storage racks and fuel assemblies were modeled as a solid block of homogeneous material. The thermophysical properties for the each material making up the rack and the assemblies were “smeared.” For each constituent of the rack “smear” temperature dependent functions for the thermal conductivity, density, and heat capacity were gathered. This of course includes the salt occupying the gaps in the assemblies. These functions were then weighted using the appropriate volume fractions. Appendix A shows the detailed calculations where this was performed. Once property functions were derived which approximated the average heat behavior inside the rack model, the expressions were entered into the material properties library in COMSOL.

The dimensions of the pool and heat load were determined somewhat arbitrarily using various references as a guide to achieve values that could be considered ‘typical.’ Appendix A also contains a table of the various information that was collected and used to define the modeled SFP specifications.

The COMSOL models were essentially analyses of the conductive and heat storage capabilities of the materials and geometry in the spent fuel pool and walls. The heat removal via the water channels is left as future work for reasons that will be discussed later. For the purpose of modeling the only heat transfer inside the SFP, the walls were initially given outward heat flux boundary conditions. A given model, heat load, and outward heat transfer area would yield a required outward heat flux that would result in a steady state condition if materials allowed. For example, if the heat load in a single rack pool was 10kW, and a total of 10m² was available for heat transfer to the adjacent water channel, the heat analysis of the pool would assume that a flux of 1kW/m² was achieved at the interface between the water and outer pool wall. Should conductive heat transfer have demonstrated the ability to remove even a fraction of the heat produced, the outward heat flux would have been reduced based on analysis of the resulting heat flux at inner surfaces. That is, if 1kW proved to pull heat away from the material faster than it

could be replaced, resulting in a temperature decrease, the outward heat flux would be reduced until a realistic balance was achieved. This analysis should still be performed with the materials suggested, though it is left as future work.

Conduction Reliant Models

It was discovered early in modeling that even conduction will be ineffective at removing the tremendous amount of heat produced in spent nuclear fuel. Early models were designed in an attempt to optimize heat removal through. In one such model the pool was designed so that the assemblies with the highest heat loads would be stored against the walls near the aluminum conduction plates. Other assemblies were tightly packed near the walls leaving as little of the poorly conductive salt in between them and the aluminum panels as possible. Figure 6 shows this model in COMSOL along with dimensions. The right boundary marks the centerline of the pool, and was given a symmetric boundary condition. The fuel rack smear discussed above was performed and applied to the fuel blocks, however separate heat loads were given to the new and old fuel regions as shown in the figure. A total SFP heat load was initially chosen to be slightly higher than the greatest heat load dealt with in the Fukushima incident. The heatload in Fukushima unit 4 SFP was 2.3MW, leading to a 2.5MW heat load modeled in this design.¹

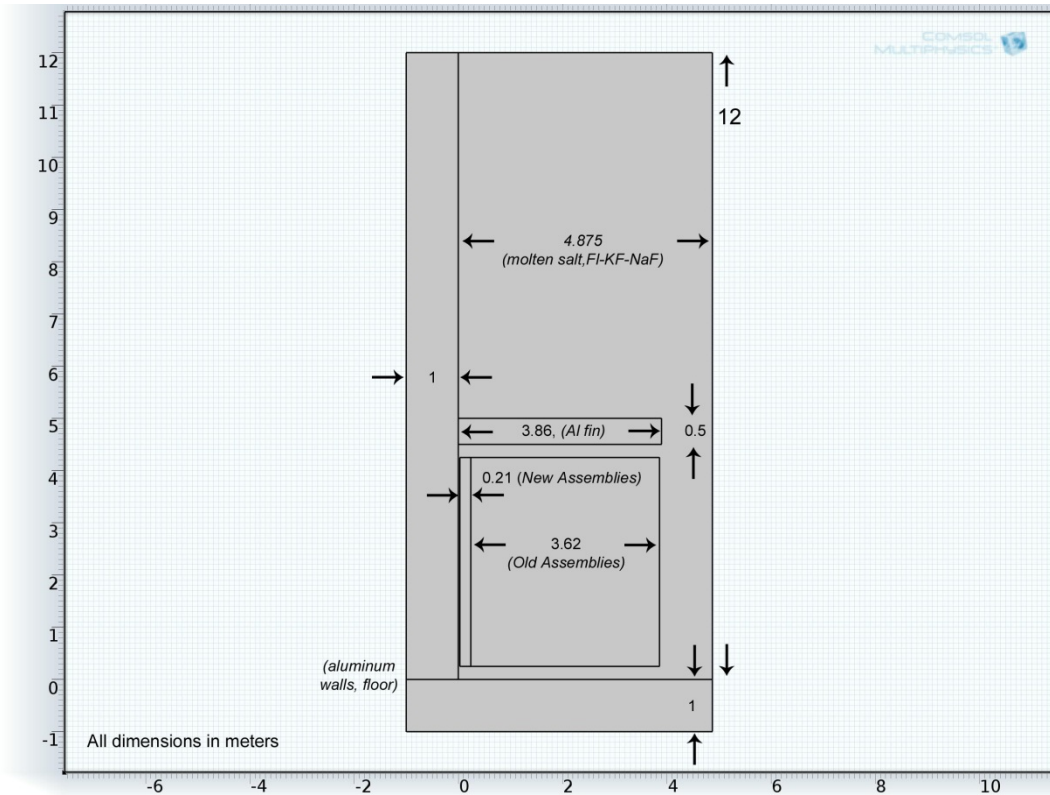


Figure 6: 2-dimensional SFP model optimized for conductive heat removal.

The fuel racks in Figure 6 are 4m tall, 8cm off the floor, and 3cm from the wall. The pool modeled is designed to hold 16 fuel racks, each 1.9m by 2.4 m, holding 99 assemblies.¹ The total pool dimensions are 9.74m squared by 12m tall. The figure allows for 3cm gaps to exist between fuel racks and between the racks and walls in one dimension, that in which the racks are 2.4m long. The other dimension allows for a larger gap in the center, shown on the right in Figure 6. The row nearest the wall (shown as the 21cm slice on the left) is occupied by 100 new assemblies. The other rack positions are occupied by 650 old assemblies.

Table 3, below shows the calculations made to determine the heat loads and outward heat flux to be applied to the model. The 2.5MW heat load was divided between the new and old assemblies as shown in the table. The total volume of the assembly/rack smear includes the 3cm gaps between adjacent racks, but not the gap between the wall. The aisle on the right in Figure 6 is not included in the smear. The volume of the new fuel block includes the outermost 21cm of the total fuel volume on all sides. The remaining volume is that of the old fuel block. The values shaded in blue in Table 3 were used as the inputs in the COMSOL model. The initial temperatures in the fuel and salt were set at 737K, 10K above the salt melting temperature.

Potential benefits gained from using a salt with a lower melting temperature are discussed in the *Future Work* section.

Table 6: 2-dimensional SFP model heat-load calculations¹

Total Heat Load	2500	kW	Old Assy Heat Load	1	kW
Total from old assys	650	kW	New Assy Heat Load	18.5	kW
Total from new assys	1850	kW	Old Assys	650	
			New Assys	100	
Total fuel/rack volume	296.9	m ³			
Old Assy volume	268.5	m ³	Old heat load per unit volume	2.42	kW/m ³
New Assy volume	28.4	m ³	New heat load per unit volume	65.04	kW/m ³
2-d slice (1m thick) calculations					
Old Assy volume	14.5	m ³			
New Assy volume	0.8	m ³			
2-d heat transfer area	12	m ²	Total heat load from old assys	35.06	kW
Outward heat flux	7.47	kW/m ²	Total heat load from new assys	54.64	kW

Figure 7, immediately following, is a plot of the temperature behavior at the fuel peak, a salt temperature near the fuel, and a salt average temperature. The fuel peak location is near the center of the old assembly block. The salt (near fuel) value is at a location approximately 3cm above the fuel at the hottest point to provide an indication of the temperature gradient above the fuel. The salt average temperature is taken in the geometrical center of the pool. A temperature limit of 1483K is used to determine when failure occurs and is plotted as a dashed line. This is the melting temperature of the Alloy-718 grid straps.⁶ Note that the fuel peak temperature would only exist at the fuel centerline, which has a much higher temperature limit. The Zircaloy cladding and Alloy-718 grid straps would exist at a lower temperature than the peak temperature reported. It is left as future work to determine the temperature delta between the fuel centerline, inner clad surface, and grid straps. It is assumed in this study that these temperatures will exist somewhere in between the fuel peak temperature and the salt temperature near the fuel. Since this latter value reaches our limit a mere half-day after the fuel peak, which violates the limit shortly after 2 days, further exploration of this model was not pursued.

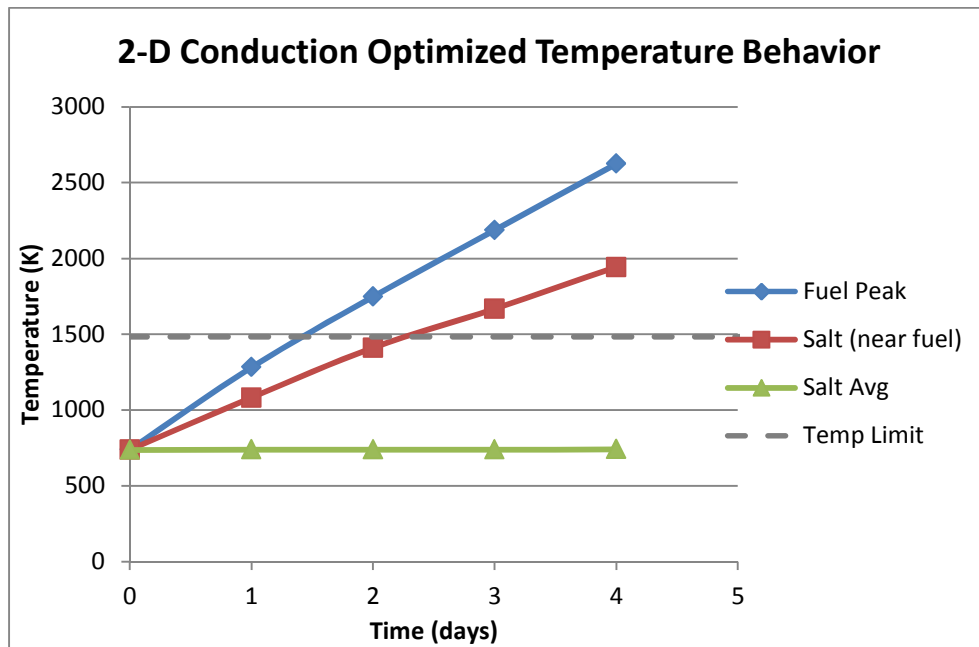


Figure 7: 2-d conduction reliant model; temperature behavior up to four days.

Figure 8 shows the graphical results produced by COMSOL at 1 day and 5 days. Of particular importance in these graphics is the negative temperatures that have appeared in the aluminum walls. This unphysical condition results from heat being drawn away from the aluminum faster than it can be replaced by the heat source. This indicates that the outward heat flux modeled is more than the aluminum can physically accomplish.

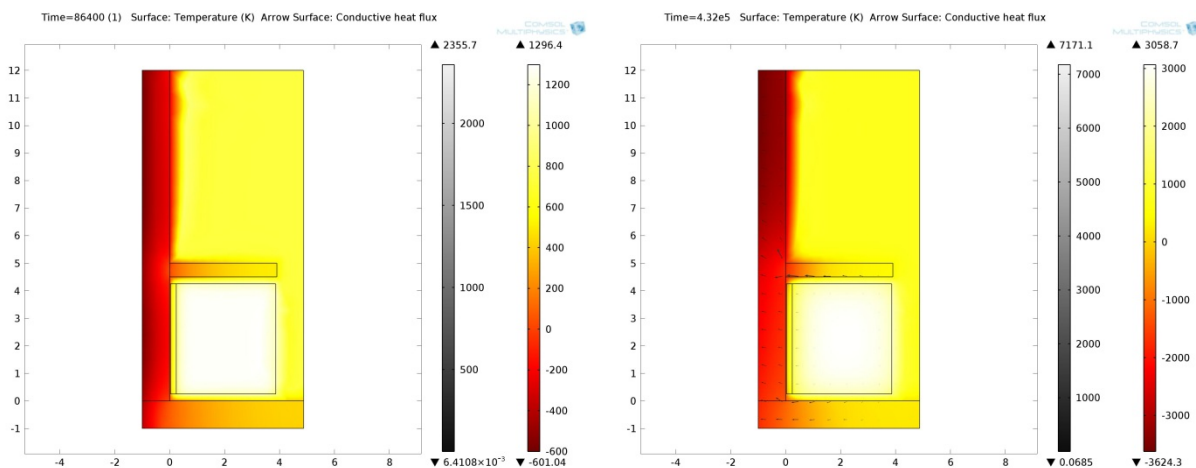


Figure 8: 2-d conduction reliant model results at 1 and 5 days.

Heat Storage Models

Concluding that conductive heat removal will not be the primary mechanism of keeping the fuel below the set temperature limits, other models were created which distributed the fuel evenly throughout a larger pool. The pool was resized to 12m squared, and the old and new assemblies were not separated as with the smaller model. The fuel/rack smear was centered in the pool, 11.5m squared by 4m tall. Various heat loads and geometries were explored. The model shown below has a total of 16 aluminum fins, each 2m wide by 0.5m thick positioned four per wall over the fuel. While a more practical model would ideally contain fewer fins, this model is shown to demonstrate the inadequacy of conduction to remove the decay heat. The entire length and thickness of the pool walls are composed of aluminum for the same reason. This more complex design was modeled in three dimensions and is shown below, in Figure 9.

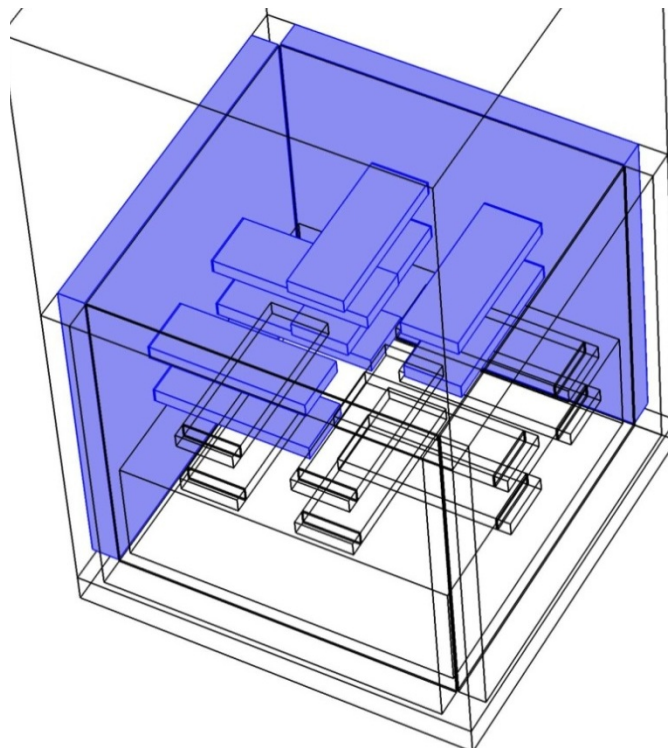


Figure 9: 3-dimensional heat storage model with outward heat flux and conduction fins.

In the particular model shown, a total heat load of 3MW was applied to the fuel. This increase in the heat load was performed to test the limits of the conductive and storage capabilities. Table 4 shows the heat load calculations used in the 3-dimensional model with outward flux.

Table 7: 3-dimensional SFP model with outward heat flux, heat-load calculations¹

Total Heat Load	3000	kW
Total fuel/rack volume	529.0	m ³
Heat load per unit volume	5.671	kW/m ³
Heat flux surface area	576	m ²
Outward heat flux	5.208	kW/m ²

The symmetry of the design was used to reduce computational demands by 8, resulting in the results shown in Figures 11 & 12. The arrows shown represent the direction and magnitude of the heat flux in the aluminum walls. The arrow color corresponds to the left-most color bar which is in units of W/m². The maximum heat flux shown exists in the aluminum fins as expected and is around 500 W/m². While this is significant, and warrants further modeling and optimization of the conductive capabilities of the design, it is clearly not sufficient to delay the fuel heat-up appreciably. Notice in Figures 10 and 11 that we are still seeing unphysically low temperatures in the aluminum walls, a manifestation of the imbalance in the heat flux applied at the outer surface and the heat flux actually achievable inside the aluminum.

Spent Nuclear Fuel Storage in a Molten Salt Pool

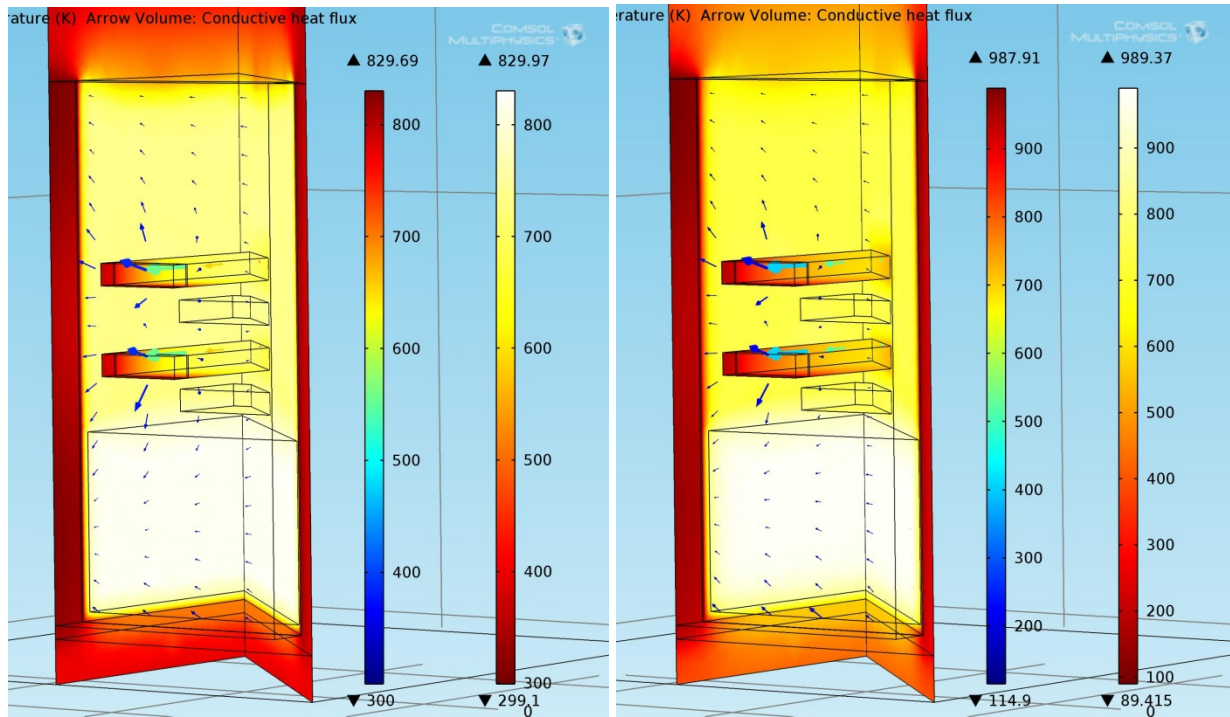


Figure 10: 3-dimensional heat storage model with outward heat flux; results at 1 day and 3 days.

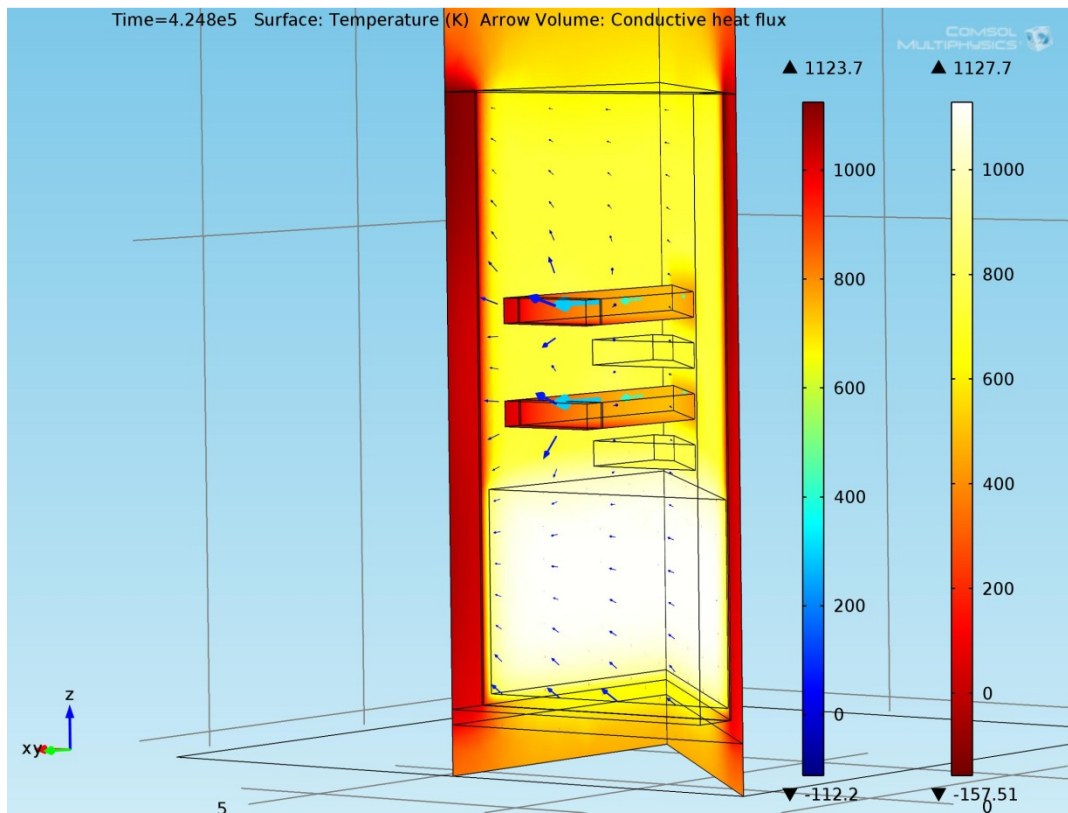


Figure 11: 3-dimensional heat storage model with outward heat flux; results at 5 days.

The temperature behavior of the 3MW heat storage model is shown below in Figure 12. Unfortunately, due to the complexity of this model, calculation beyond five days was not possible with the resources available. However, the temperature behavior of the fuel centerline and the near-fuel salt temperature are decidedly linear. These extrapolations show the fuel and salt temperatures violating the limits at just over nine days and fifteen days, respectively. As mentioned previously, we would expect the true behavior at the clad and grid straps to lie somewhere between these two values.

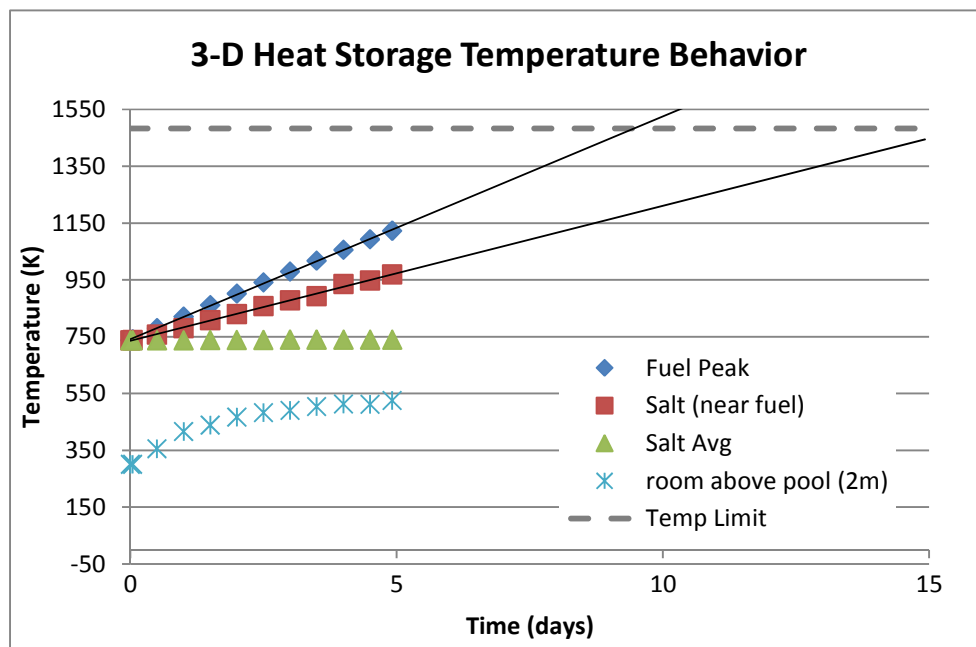


Figure 12: 3-dimensional heat storage model with outward heat flux; temperature behavior up to five days.

A final calculation was performed in which the outward heat flux was eliminated, along with the conduction fins. This was an attempt to reduce computation demand as well as investigate the perceived irrelevance of the conduction fins and water channels. The heat load in this model was also reduced to 2MW; a specification that would be considered an operations limit. The COMSOL input heat load was thus 3.781kW/m^3 .

The COMSOL results of the 3-dimensional heat storage model with no heat flux at 1, 5, 10, and 15 days are shown in Figures 13 and 14. With the heat flux removed and the total heat load reduced, modeling out to the temperature limit violation is possible. Notice that the aluminum

temperatures remain below the melting temperatures despite the insulated boundary conditions. Notice also the tremendous heat storage capabilities of the salt, evidenced by its reluctance to heat-up.

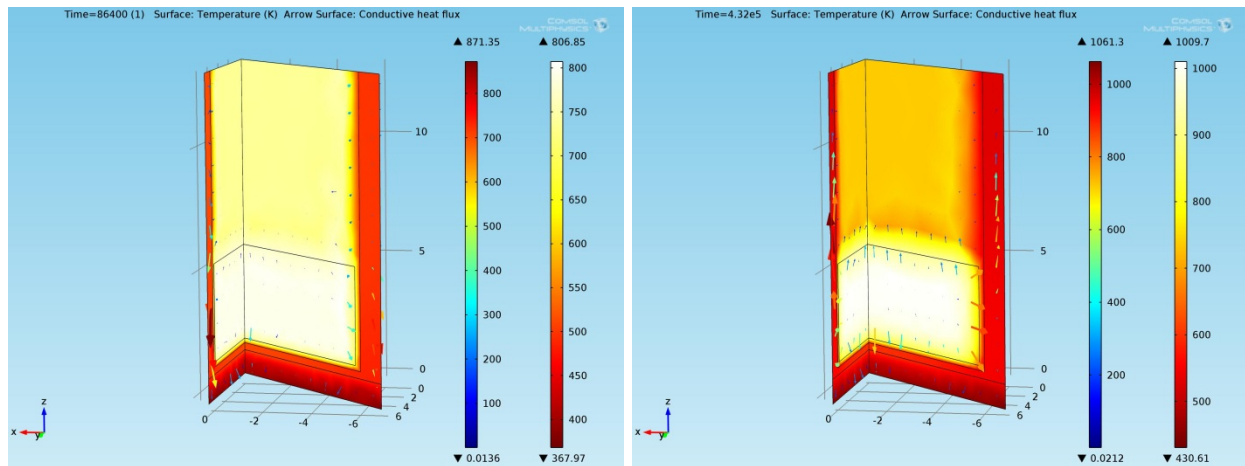


Figure 13: 3-dimensional heat storage model with no outward heat flux; results at 1 & 5 days.

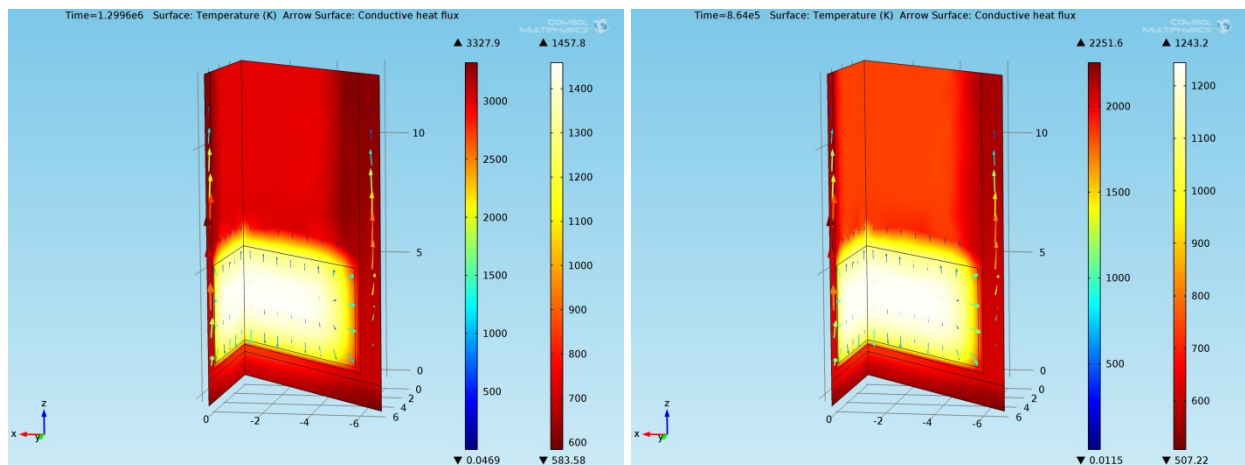


Figure 14: 3-dimensional heat storage model with no outward heat flux; results at 10 & 15 days.

Figure 15, below, shows the transient temperature behavior of the 3-d insulated 2MW model. The fuel peak temperature reaches the proposed limit on the 16th day after offsite power loss has been simulated. The extrapolation of the high salt temperature does not violate the limit until day 25.

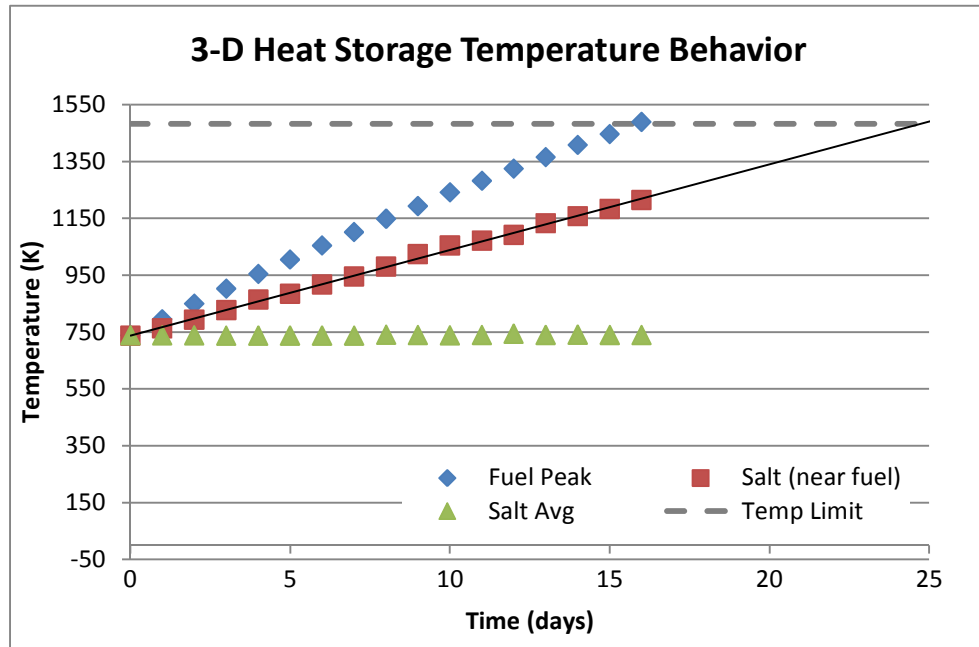


Figure 15: 3-dimensional heat storage model with no outward heat flux; temperature behavior up to sixteen days.

Future Work

While the thermal analysis of the molten salt fuel pool has been very promising there is work yet to do in order for this concept to be developed. Outside of the subject of heat removal and storage, there are many logistics to consider. A thorough investigation of potential salts to occupy the pool is in order and must be accompanied by a detailed corrosivity analysis. FLiNaK was used in the model shown because it can be assumed that corrosivity will not be an issue given the salts original design intent, and because of its low neutron moderation and high thermal conductivity. Other salts may exist that will serve these concerns as well without the cost of such a high melting temperature.

Detailed analysis of the normal operation salt cooling loop and heat exchanger to determine flow rates, pipe sizes, materials, and salt heaters to reheat the salt after flow has stopped in an accident scenario. Should FLiNaK be chosen, investigation of the extent of HF generation in the intermediary washing chamber must be performed, the generation addressed.

Regarding the thermal analysis begun in this study, there are steps that the authors would have liked to address, but were unable to given the time constraints of the project. The most relevant are listed here.

- The heat storage models should be optimized to determine the time frame before temperature violation at various heat loads and conduction configurations.
- Fuel assemblies and racks should be modeled in detail to assess the validity of the fuel/rack smear, improve the smear characteristics, and determine cladding and grid temperatures as a function of the peak temperature in the fuel/rack smear.
- The natural circulation of the emergency cooling channels must be modeled in detail to determine the optimal channel configuration and the heat transfer achievable at the pool/water interface.

Finally, with the preceding pieces accomplished, a new analysis could be performed utilizing an accurate outward heat flux, an optimal SFP geometry, and knowledge of the true material temperatures based on the peak temperature. Given that we have not taken credit for any outward heat flux, nor for the temperature delta between the peak temperature and the fuel structure temperatures, such an analysis could potentially yield an improved time-to-melt that eclipses the results already achieved.

Conclusions

We have described a spent fuel pool design that utilizes a molten salt instead of water which would surround and flow through the fuel, removing decay heat in the same way water does in currently operating pools. An emergency response system has been suggested which implements a naturally circulating water loop surrounding the pool and aiding in heat removal after a loss of off-site power accident occurs. Most importantly, this study has demonstrated that even with no means of heat removal, a spent fuel pool with a total heat load of 2MW can be kept under the lowest melting temperature for over 15 days via heat passive transfer to the salt. Where water cannot accomplish this due to boiling at 100C, the molten salt will remain covering the fuel for weeks even after an unspeakable spent fuel melt disaster. Further analysis discussed previously

would boost this length of time by providing accurate knowledge of the heat removal through emergency water systems.

Using a molten salt has several other benefits beyond superior heat absorption. It is not a neutron moderator, as is water, totally eliminating criticality concerns in the spent fuel pool. It is denser than water, providing greater gamma shielding possibly reducing the required depth of the pool. Finally, because the fuel will be surrounded by molten salt and not water, there will be no hydrogen-producing oxidation reactions which have been the cause of so much trouble at Fukushima.

It is the recommendation by the authors that this concept be investigated and tested. The difficulties of working with a molten salt in the SFP are readily apparent, but so are the benefits. A thorough cost/benefit analysis is in order, along with more detailed studies that address the concerns noted in Future Work. That said, the potential benefits of this concept are great in terms of Beyond Design Basis preparedness.

References

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- 7 COMSOLTM Multiphysics [4.2a] 1 New England Executive Park Suite 350 Burlington, MA 01803 USA
- 8 Sohal, Manohar, Matthias Ebner, Phil Sharpe, and Piyush Sabharwal. United States. U.S. Department of Energy National Laboratory Idaho National Laboratory. *Engineering Database of Liquid Salt*

In order to model the heat load produced in the SFP, it was necessary to create a homogeneous material to stand in for the heterogeneous rack and assembly configurations. From Reference 104, the masses of a typical PWR assembly constituents are listed in Table A1. The densities are calculated from functions shown in Table A2 at 770 K, the probably normal operating temperature of the SFP. From the densities and a volume of 19.38m^3 , a volume that must be filled by the molten salt in order to form the solid cell block was calculated. A salt mass was then calculated.

It is predicted that the fraction by volume of the rack cell constituents would be the most accurate weighting function to use in assembling a “smeared” set of property functions. This is because the required distance through each material to be traversed is better reflected in the volume fractions than in the mass fractions. The volume fractions are calculated in the 5th column in the table below.

Table A1: Rack smear weighting function calculations^{6,7,8}

Rack Model Material	Mass per Assy(kg)	Density (kg/m ³)	Volume (m ³)	Volume fraction	Weight fraction
Salt	266.461	2094.5	1.27E-01	6.49E-01	2.86E-01
SS 304L (rack)	0.089	7652.2	1.16E-05	5.91E-05	9.51E-05
Zircaloy-4	125.600	6558.4	1.92E-02	9.77E-02	1.35E-01
SS 304L (assy)	16.400	7652.2	2.14E-03	1.09E-02	1.76E-02
UO ₂	521.200	11035.5	4.72E-02	2.41E-01	5.59E-01
Inconel 718	2.1000	8059.1	2.61E-04	1.33E-03	2.25E-03
Sum	9.32E+02		1.96E-01	1.00E+00	1.00

The respective coefficients were each weighted by the material’s volume fraction and then added, resulting in a single parametric expression for each property for the smeared rack material. Table A2, below shows the parent expressions and the final smeared expressions. Of special interest is the column titled “Non-simple term.” The two terms that populate these cells in the table were $[225000 \cdot \text{EXP}(-12410/T)/T]$, and $[-0.9285 \cdot (T-3120)]$, respectively. In the end the exponential term was dropped from the uranium conductivity expression because it did not contribute significantly at the temperatures of interest and presented problems when attempting to load the expressions into COMSOL.

Table A2: Design Inputs

Fuel Assembly Materials & Dimensions	
	Mass (kg)
SS 304L	16.4
Zircaloy-4	125.6
Alloy 718	2.1
UO ₂	521.2
Ref: [104]	
Storage Rack Material & Dimensions	
Assembly array: 9 by 11	
Width: 2.37m	
Length: 1.94m	
Height (pool floor to rack top): 4.06m	
Total rack cell volume (including 4cm inter-rack gap): 19.38 m ³	
Rack material: SS 304L (density = 8027.2 kg/m ³)	
Rack mass: 8772.9kg	
Typical SFP capacity: 2000 assemblies	
Typical SFP dimensions: 30ft by 40ft by 40ft deep	
Gap of 3.5 - 7.5 inches between walls and racks	
Ref: [104]	
Heat Load Design Inputs	
Fukushima SFP 4 Heatload: 2.3 MW	
Fukushima assembly loading (BWR assemblies):	
1331 used; 204 new	
Ref: [105]	

Table A3: Temperature dependent properties and weighted homogenization functions

T 777 K			Property Fit Terms									Value at Set Temp
Reference	Material	Weight Factor (volume fraction)	Property	T ⁰	T ¹	T ²	T ³	T ⁴	T ⁵	T ²	Non-simple term*	
103	LiF-NaF-Kf	0.649	k	0.36	5.60E-04							0.80 W/m*K
103			p	2579.3	-0.624							2094.45 kg/m ³
103			C	976.78	1.0634							1803.04 J/kg*K
101	Zircaloy-4	0.0977	k	12.767	-5.43E-04	8.98E-06						17.77 W/m*K
101			p	6673.019	-0.1474608							6558.44 kg/m ³
101			C	261.4449	0.08689762							328.96 J/kg*K
101	SS 304 L	0.0110	k	6.742253	0.02864915							29.00 W/m*K
101			p	7945.333	-0.1981948	-3.71E-04	2.21E-07	-5.13E-11				7652.25 kg/m ³
101			C	109.2073	2.571775	-0.006528099	7.78752E-06	-4.168E-09	8.09061E-13			529.36 J/kg*K
102	UO2	0.241	k	2.3							3.35E-05	2.30 W/m*K
102			p	8860							2175.4755	11035.48 kg/m ³
102			C	52.1743	8.80E-02	-8.42E-05	3.15E-08	-2.63E-12		-7.14E+05		82.31 J/kg*K
101	Inc 718	0.00133	k	3.495867	0.02673305	-1.12E-05	3.61E-09	8.24E-14				19.24 W/m*K
101			p	8288.965	-0.05548692	-7.19E-04	7.97E-07	-3.48E-10				8059.09 kg/m ³
101			C	361.3373	0.2378248	7.56E-06						550.69 J/kg*K

SMEARED TEMP DEPENDENT PROPERTIES

k	2.1139E+00	6.6083E-04	8.6266E-07	4.7947E-12	1.0948E-16	0.0000E+00	0.0000E+00	8.0772E-06	3.15	W/m*K
p	4.5591E+03	-4.2166E-01	-5.0382E-06	3.4929E-09	-1.0261E-12	0.0000E+00	0.0000E+00	5.2417E+02	4753.90	kg/m ³
C	6.7376E+02	7.4845E-01	-9.2048E-05	9.3205E-08	-4.6451E-11	8.8937E-15	-1.7201E+05	0.0000E+00	1226.24	J/kg*K

** Will have to manually add last column in order to preserve Temp dependency

Appendix B: MCNP Criticality Analysis

Input file for pool filled with water at 100°C

Message: Datapath=C:\MCNP\mcnp5

Senior Design Project Criticality Calculation - Water, 373K

```
c
c cell cards
c
c unit fuel pin cell
1 4 -10.52 -2 u=1
3 3 -6.55 +2 -3 u=1
4 1 -1.0 +3 u=1
c unit instrument tube cell
5 1 -1.0 -4 u=2
6 3 -6.55 +4 -5 u=2
7 1 -1.0 +5 u=2
c pin cell mesh
17 1 -1.0 -14 +15 -16 +17 u=4 lat=1
    fill= -9:9 -9:9 0:0
        4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 2 1 1 2 1 1 2 1 1 1 1 4
        4 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
c unit fuel assembly
18 1 -1.0 -18 +19 -20 +21 -22 +23 fill=4
c the abyss
999 0 (+18:-19:+20:-21:+22:-23)

c
c surface cards
c
c cylindrical surfaces of a fuel pin cell
1 cz .412
2 cz .412
3 cz .476
c cylindrical surfaces of an instrument tube
4 cz .570
5 cz .610
c surfaces of the unit pin cell
14 px .6325
15 px -.6325
16 py .6325
17 py -.6325
c surfaces of the unit fuel assembly
18* px 10.75
19* px -10.75
20* py 10.75
21* py -10.75
22* pz 182.88
23* pz -182.88

c
c data cards
c
TMP 3.2154e-8 8r
IMP:n 1 7r 0
c =====
```

Appendix B: MCNP Criticality Analysis

```
c ==MODERATOR at Oppm=====
m1 1001.71c -1.1190E-01
    8016.71c -8.8810E-01
mt1 lwtr.10t
c ==FLiNaK=====
c m1 3007.71c -1
c 11023.71c -1
c 19039.71c -1
c 9019.71c -3
c ==CLADDING=====
m3 40000.58c -98.18
    50116.71c -1.4
    24052.71c -0.1
    26056.71c -0.2
    8016.66c -0.12
c ==FUEL, 5.0%=====
m4 92235.71c +1.2E+21
    92234.71c +9.5E+18
    92238.71c +2.2E+22
    8016.71c +4.7E+22
c =====
kcode 4000 1.0 50 1000
sdef x=d1 y=d2 z=d3
spl 0 1
sil -10.74 +10.74
sp2 0 1
si2 -10.74 +10.74
sp3 0 1
si3 -182.78 +182.78
```

Input file for pool filled with salt at 100°C

Message: Datapath=C:\MCNP\mcnp5

Message: Datapath=C:\MCNP\mcnp5

Senior Design Project Criticality Calculation - FLiNaK, 373K

```
c
c cell cards
c
c unit fuel pin cell
1 4 -10.52 -2 u=1
3 3 -6.55 +2 -3 u=1
4 1 -2.097 +3 u=1
c unit instrument tube cell
5 1 -2.097 -4 u=2
6 3 -6.55 +4 -5 u=2
7 1 -2.097 +5 u=2
c pin cell mesh
17 1 -2.097 -14 +15 -16 +17 u=4 lat=1
    fill= -9:9 -9:9 0:0
        4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 2 1 1 2 1 1 2 1 1 1 1 4
        4 1 1 1 2 1 1 1 1 1 1 1 1 1 2 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
c unit fuel assembly
18 1 -2.097 -18 +19 -20 +21 -22 +23 fill=4
c the abyss
```

Appendix B: MCNP Criticality Analysis

999 0 (+18:-19:+20:-21:+22:-23)

```
c
c surface cards
c
c cylindrical surfaces of a fuel pin cell
1  cz      .412
2  cz      .412
3  cz      .476
c cylindrical surfaces of an instrument tube
4  cz      .570
5  cz      .610
c surfaces of the unit pin cell
14 px      .6325
15 px     -.6325
16 py      .6325
17 py     -.6325
c surfaces of the unit fuel assembly
18* px      10.75
19* px     -10.75
20* py      10.75
21* py     -10.75
22* pz      182.88
23* pz     -182.88

c
c data cards
c
TMP      3.2154e-8 8r
IMP:n    1 7r 0
c =====
c ==MODERATOR at 0ppm=====
c m1      1001.71c -1.1190E-01
c          8016.71c -8.8810E-01
c mt1 lwtr.71
c ==FLiNaK=====
m1       3007.71c  -1
          11023.71c -1
          19039.71c -1
          9019.71c  -3
c ==CLADDING=====
m3       40000.58c -98.18
          50116.71c -1.4
          24052.71c -0.1
          26056.71c -0.2
          8016.66c  -0.12
c ==FUEL, 5.0%=====
m4       92235.71c +1.2E+21
          92234.71c +9.5E+18
          92238.71c +2.2E+22
          8016.71c +4.7E+22
c =====
kcode 4000 1.0 50 1000
sdef x=d1 y=d2 z=d3
spl 0 1
si1 -10.74 +10.74
sp2 0 1
si2 -10.74 +10.74
sp3 0 1
si3 -182.78 +182.78
Input file for pool filled with salt at 500°C
```

Message: Datapath=C:\MCNP\mcnp5

Senior Design Project Criticality Calculation - FLiNaK, 773K

```
c
c cell cards
c
c unit fuel pin cell
1  4  -10.52      -2      u=1
3  3   -6.55      +2     -3  u=1
4  1  -2.097      +3      u=1
c unit instrument tube cell
```

Appendix B: MCNP Criticality Analysis

```

5  1  -2.097      -4      u=2
6  3  -6.55       +4  -5    u=2
7  1  -2.097      +5      u=2
c  pin cell mesh
17 1  -2.097      -14 +15 -16 +17  u=4 lat=1
    fill= -9:9 -9:9 0:0
        4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 2 1 1 2 1 1 2 1 1 1 1 4
        4 1 1 1 2 1 1 1 1 1 1 1 1 2 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 2 1 1 2 1 1 2 1 1 2 1 1 2 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 2 1 1 1 1 1 1 1 1 2 1 1 1 4
        4 1 1 1 1 1 2 1 1 2 1 1 2 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4
        4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
c  unit fuel assembly
18 1  -2.097      -18 +19 -20 +21 -22 +23 fill=4
c  the abyss
999 0  (+18:-19:+20:-21:+22:-23)

c
c  surface cards
c
c  cylindrical surfaces of a fuel pin cell
1  cz      .412
2  cz      .412
3  cz      .476
c  cylindrical surfaces of an instrument tube
4  cz      .570
5  cz      .610
c  surfaces of the unit pin cell
14 px      .6325
15 px      -.6325
16 py      .6325
17 py      -.6325
c  surfaces of the unit fuel assembly
18* px      10.75
19* px      -10.75
20* py      10.75
21* py      -10.75
22* pz      182.88
23* pz      -182.88

c
c  data cards
c
TMP  6.6622e-8 8r
IMP:n  1 7r 0
c  =====
c  ==MODERATOR at 0ppm=====
c  m1      1001.71c -1.1190E-01
c          8016.71c -8.8810E-01
c  mt1 lwtr.71
c  ==FLiNaK=====
m1  3007.71c  -1
    11023.71c -1
    19039.71c -1
    9019.71c  -3
c  ==CLADDING=====
m3  40000.58c -98.18
    50116.71c -1.4
    24052.71c -0.1
    26056.71c -0.2

```

Appendix B: MCNP Criticality Analysis

```
      8016.66c  -0.12
c      ==FUEL, 5.0%=====
m4     92235.71c +1.2E+21
      92234.71c +9.5E+18
      92238.71c +2.2E+22
      8016.71c  +4.7E+22
c      =====
kcode  4000 1.0 50 1000
sdef   x=d1 y=d2 z=d3
spl    0 1
sil    -10.74 +10.74
sp2    0 1
si2    -10.74 +10.74
sp3    0 1
si3    -182.78 +182.78
```