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Evaluate the Adaptability of Present Licensing Requirements to New Nuclear Fuels for Use in Current Reactors

By

Samuel A. Paletta

Thesis submitted to The College of Engineering and Mineral Resources
At West Virginia University
in partial fulfillment of the requirements
for the degree of

Master of Science
in
Engineering

Approved by

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Chemical Engineering

Morgantown, West Virginia
2005

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Abstract

Evaluate the Adaptability of Present Licensing Requirements to New Nuclear Fuels for Use in Current Reactors

By Samuel A. Paletta

Currently there is no comprehensive plan available for the methodology, timeline, and cost structure needed when licensing for new nuclear fuels, cladding, and geometry. By combining various sources from the NRC, nuclear vendor documents, and experience from engineers who have worked on licensing standard nuclear materials, a methodology, timeline, and cost structure for licensing new fuels, cladding, and geometry was developed. The specific scenarios of fuels, cladding, and geometry were used to outline the common advantages and disadvantages of changing the physical properties or physical structure of the standard nuclear fuel assembly. From these changes, the timeline and costing analyses were performed. It was determined from the timeline and costing analyses that the most advantageous development scenario was to combine the fuel and cladding licensing together and license the fuel/cladding combination for over 100 Gigawatt-days per metric ton of uranium.

Dedication

To Erin and Andrew
my rock and my reason
I love you both

Acknowledgement

I would first like to thank my wife, Erin, for always being there when I needed her and making me a better person.

Next, I would like to thank Dr. Al Stiller who took a young BS chemist under his wing and turned him into a chemical engineer. He was always there to listen and dole out advice not just in engineering but in life. I owe a lot of where I am now to him.

I would also like to thank Ed Lahoda for giving me a chance at Westinghouse. He allowed me to take my projects at Westinghouse and use them for my Master's thesis. He made all of the technical portions of this paper better through his knowledge and expertise within the nuclear industry.

To my brothers Rich and John, and to my sisters Maria and Nicole, you guys have helped me throughout my life and I know that you will always be there for our family. Thanks for everything.

Finally, I would like to thank my parents. They have endured my seven and a half years of college to finally see their son get his degree. Even though they joked and made fun with me about how long it took, they always were supportive. I hope that I make you both proud.

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Chapter 1 – Introduction

Nuclear energy currently accounts for approximately 20% of the electricity generated in the United States. The rest is manufactured with mostly coal with oil, natural gas, and renewable energy sources making up a much smaller portion. There major difference between a commercial nuclear power plant and the same size coal plant is the fuel utilized to turn liquid water into steam. They both use a Rankine cycle to make electricity. A Rankine cycle consists of a boiler to turn pressurized water into steam, a turbine where the steam's energy is extracted to a generator by decreasing its pressure, a condenser to cool the steam back into liquid water and finally a pump to increase the pressure of the liquid water back to its original pressure before again entering the boiler.

The commercial nuclear plant is exceedingly regulated in comparison to the other sources of electricity. This is understandable since the other methods to make electricity do not have the possibility of causing surrounding areas to become uninhabitable for centuries. Because of the dangers associated with nuclear energy, the federal government has always had a role in the way it is produced and researched.

The first United States government agency in charge of commercial nuclear energy was the Atomic Energy Commission (AEC). The AEC was founded on August 1st, 1946 by the Atomic Energy Act. However, not until 1954, when the Atomic Energy Act was revised to permit commercial use of nuclear energy to produce electricity as well as regulate its safety, was the commercial use of nuclear energy actually possible. The AEC was in control of both the weapons use and commercial use of nuclear power. Because of a perceived conflict of interest between production and safety, in 1974 the AEC was dissolved and two new government agencies were created. The Nuclear Regulatory Commission was created to control and regulate the commercial aspects of the nuclear power industry. The second agency, the Energy Research and Development Agency (disbanded in 1977 upon creation of the Department of Energy) was created to provide research assistance for nuclear power production. [1]

[1] "**Atomic Energy Commission**." Encyclopædia Britannica. 2005. Encyclopædia Britannica Premium Service. 10 Oct. 2005 <**http://www.britannica.com/eb/article-9010129**>.

The fuel used in nuclear energy is one of the concerns of the Nuclear Regulatory Commission (NRC). Their concern is not just in the quality of the fuel, but also in protecting it from harm during normal and accident conditions, and in preventing the release of harmful levels of radioactivity to the public. These latter concerns include the entire primary system and containment structure. Using a pressurized water reactor as the baseline, the definition of the primary system is all that is in contact with the coolant around the core. The secondary system is the Rankine steam cycle. Here the boiler is a large heat exchanger known as a 'steam generator' that has primary coolant on the tube side and secondary water on the shell side. There is a third loop known as the condensate loop. This loop contains the cooling water that passes on the tube side of the condenser and is used to condense the steam back to water. Again, the secondary and condensate systems are similar to those found in any large commercial coal plant.

The core within the primary system is the main source of heat for the primary side water that heats the steam generators. The core is made up of fuel bundles or fuel assemblies. These fuel assemblies consist of a large number of fuel rods arranged in a geometric pattern so that the neutrons given off by their nuclear fission can bombard nearby fuel rods and continue with the chain reaction. Equation >>> illustrates neutron capture and the subsequent fission of uranium 235. U-235 is the main fissionable isotope of uranium, and all US commercial reactors rely on this as their source of fuel. Natural uranium is only 0.72% U-235 and must be enriched for a light water reactor (LWR) to achieve criticality. Fuels enriched up to 20% U-235 are known as low enriched uranium or LEU. However, the maximum U-235 concentration used for fuel used in commercial light water reactors is about 5%. Higher enrichments (5-100%) are used in research reactors, gas-cooled reactors, and in nuclear weapons production.

Once the uranium is enriched it is converted from UF_6 to UO_2. Uranium dioxide is the material of choice from which to construct fuel pellets. This standard has been used since the beginning of the commercial reactor in 1954. These fuel pellets are pressed and sintered until they reach a normally >95% of their theoretical density. Upon completion, they are loaded into a tube (cladding) and sealed with end plugs. Once the fuel has been loaded into the cladding and sealed, it is now known as a fuel rod. Many fuel rods are then placed within a gridded structure known as a fuel assembly. Once the fuel rods have been loaded within the fuel assembly, the fuel is ready for irradiation inside a commercial reactor. The fuel assembly or more precisely the fuel, cladding, and geometry that make up the fuel assembly are closely regulated by the NRC because it is the first line of defense in preventing the release of fission products into the environment. They fall under a set of regulations that explicitly name the requirements that the fuels must meet.

Currently, the requirements for nuclear fuels are written with the assumptions that the fuel is uranium dioxide (UO_2), that the cladding is zircaloy or ZIRLO (a zirconium based alloy), and that the overall fuel shape is cylindrical held within gridded columns. Figure 1 illustrates the general overview of a standard pressurized water reactor (PWR) fuel assembly. The fuel rod is made up of a UO_2 pellet surrounded by a zirconium alloy cladding, with welded end plugs to make it hermetically sealed. The rods are then placed into the grids for the fuel assembly and suspended above the bottom nozzle and below the top nozzle by the grid assemblies. The grid assemblies are rigid zirconium alloy squares with generally either15 x 15, or 17 x 17 cells that hold the individual fuel rods.

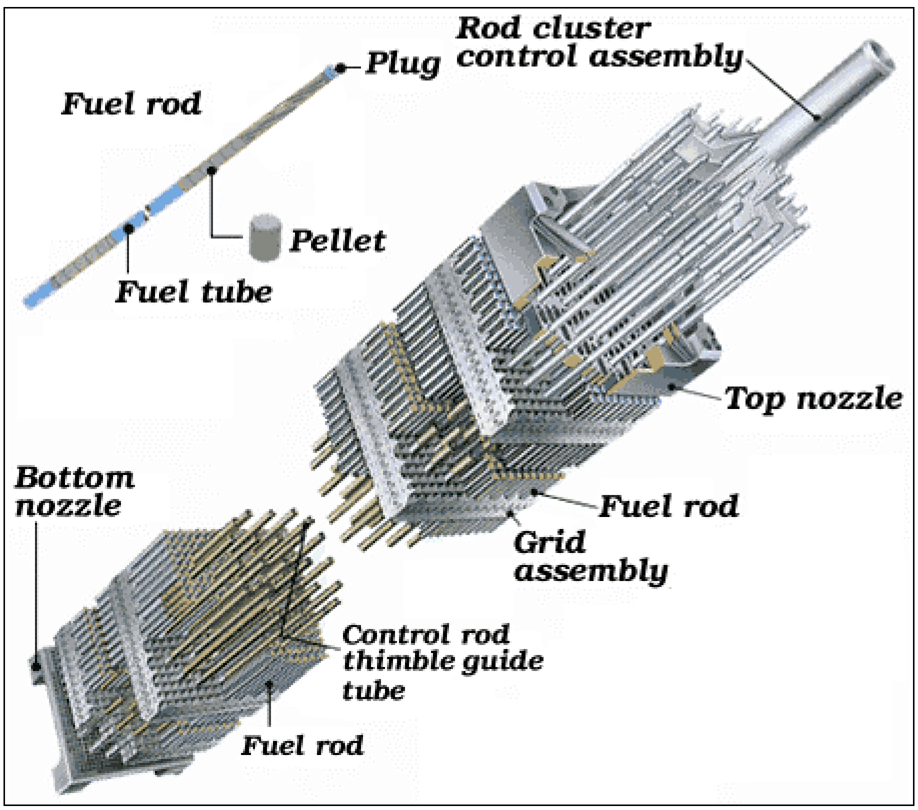

Figure 1 - Fuel Assembly Overview (modified by author) - www.mnf.co.jp/images2/2img_pwr.gif

The fuel assembly shown in Figure 1 is currently licensed for use in pressurized water reactors. To be considered 'licensed', the fuel assembly has undergone a rigid review. This review involves the models and analyses used to quantify all of the possible scenarios which the fuel assembly will be subjected. To be acceptable, the models and analyses must provide that the fuel assembly will be able to withstand both normal and accident conditions.

Because the NRC is the regulator of the commercial nuclear power industry, it has established a basic approach to how it conducts operations. The first is to develop regulations and guidance. Second, they control licensing, decommissioning, and certification. Third, they have oversight of all they regulate and license. Finally, they use operational experience to set/confirm their decisions for their regulations and guidance. All four of these are further researched, evaluated, and ruled upon. Figure 2 was taken from the NRC website (www.nrc.gov) and illustrates the cyclical process that the NRC uses to develop and enforce their rules and regulations.

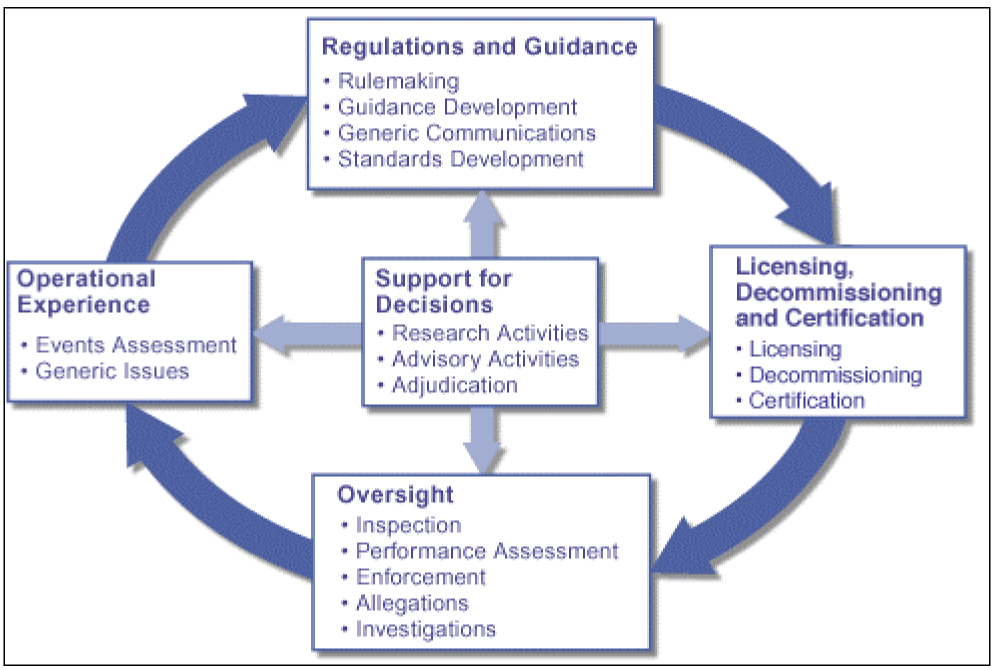

Figure 2 – "How We Regulate"; NRC website: www.nrc.gov/what-we-do/regulatory.html

When analyzing the licensing requirements for a new nuclear fuel, it is imperative to first analyze the rules/regulations that are currently in place. The current rules/regulations applicable to nuclear fuels are found in the Code of Federal Regulations, Section 10. Section 10 deals with energy, not just nuclear, and the specific subsections will be addressed at a later point. However, the setup for a fuel assembly that was previously discussed has its roots in Section 10. In order to further the understanding of the licensing process and approach that will be described in a later section, first, a general overview of the fuel, fuel rod, and fuel assembly needs to be addressed. A brief set of the requirements are laid out below. Chapter 2 – Literature Review contains the rest of the requirements and provides a more in depth review of the regulations and guidelines governing nuclear fuel, cladding and geometry. Chapter 3 contains the methodology for licensing new fuels, cladding, or geometry changes to the standard fuel assembly.

The standard fuel rod is designed to operate at the 650°F heat and the 2200 psi pressure in the interior of a pressurized water reactor. These temperatures and pressures are for the standard pressurized water reactor, and do not reflect the conditions within a boiling water reactor. The fuel rod consists of uranium fuel shaped into cylinders that are stacked and surrounded by a zirconium alloy cladding. At both ends of the fuel rod there are plugs that are welded into place. Within the fuel rod, there is a plenum that is used to capture and hold fission gases that are released during burnup. Within the plenum is a spring that holds the pellets down and keeps them from experiencing stress during normal transportation from the fuel fabrication facility to the commercial reactor. The fuel itself is chamfered and dished. The chamfering and dishing are performed to help the pellet to densify and swell uniformly during irradiation and to prevent the ends of the fuel pellet from becoming egg shaped and distorting or denting the cladding.

When the fuel is first irradiated it decreases in volume before beginning to swell. There is an initial gap that is set by the manufacturer between the cladding and the fuel pellet. This gap initially increases in width during densification but, upon swelling, the gap begins to close until approximately 20-30 GWD/MTU (Gigawatt days / metric ton of uranium) when at which time the gap is no longer open and the pellet and cladding are now in contact.

During the initial irradiation, the fuel pellets, begin to increase in density. The fuel pellets are only at a maximum of ~95% to 97% of the theoretical density when they are loaded into the fuel assemblies. This occurs because the sintering process that is done prior to placement of the pellets within the fuel rod, does not compress the pellets to their theoretical density. Then upon irradiation the resulting high temperature causes them to further densify until they reach approximately 100% theoretical density. The time it takes for this to complete is approximately 10-15 GWD/MTU.

After the pellet reaches its maximum densification, it begins to swell due to the release of fission gas products from inside the pellet. These fission gasses begin to force the pellet to expand at the same time, the fuel clad undergoes creep down due to the pressure differential between the outside of the rod (~2250 psia) and inside of the rod (~100 to 500 psia). During this time, a phenomena known as pellet cladding interaction or PCI occurs. This occurs when the pellet expands and the cladding creeps down until the gap between the pellet and the cladding is closed. Then, again at approximately 20-30 GWD/MTU, the pellet begins to exert pressure upon the cladding from the inside. Also, some of the fission gasses that were trapped inside of the pellet begin to escape and they migrate to the plenum. The fission gasses begin to exert a pressure outwards and the internal pressure of the rod begins to increase. This is measured as the rod internal pressure (RIP). The rod internal pressure is kept to less than system pressure. The reason is to ensure that the gap between the pellet and cladding does not reopen. If the gap were to reopen a situation known as departure from nucleate boiling (DNB) propagation could occur. This is caused when the gap between the pellet and cladding opens and the fuel rod balloons outward. This restricts the flow of coolant through the channels between the fuel rods and increases the possibility that the adjacent fuel rods could also balloon outward. If this local heating causes adjacent fuel rods to balloon outward, then this scenario could continue or propagate until a significant number of fuel rods have their coolant flow limited or extinguished. This can lead to localized hotspots and melting of cladding and fuel.

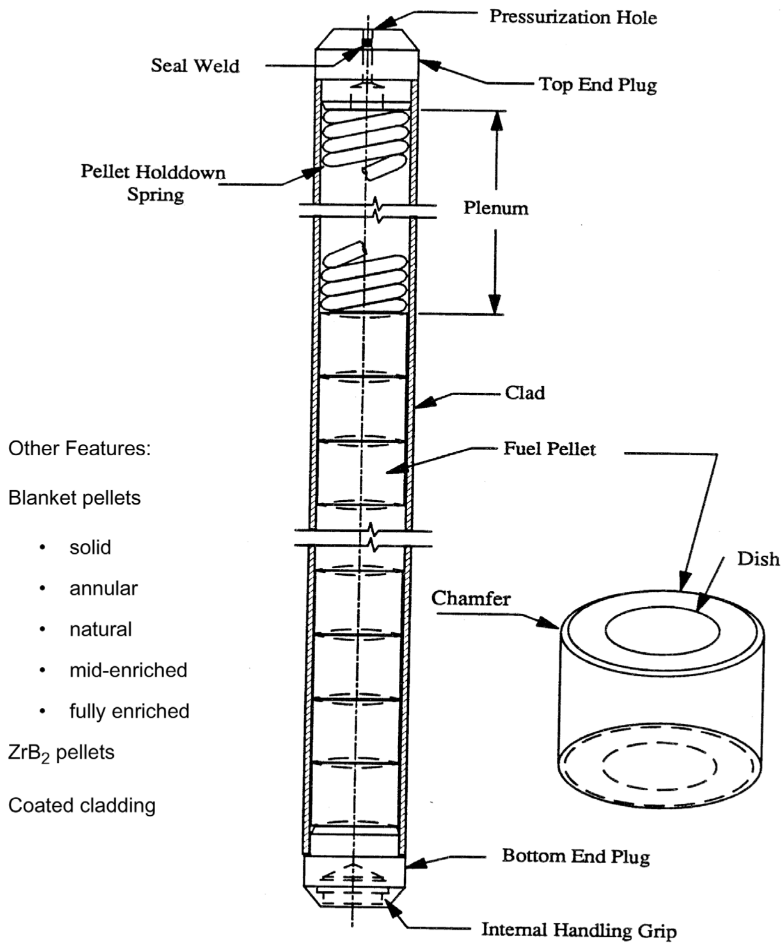

Figure 3 - Standard Westinghouse Fuel Rod

The background for licensing and fuel rod and fuel assembly construction has been addressed. Now a description of the specific licensing approaches and their timeline and cost structures will be addressed.

The approach to determining the licensing needs was determined from sources such as various documents from the NRC, vendor documents, and experience from engineers who have worked on licensing. The above sources allowed for a determination of the needs for each three different licensing scenarios. Specifically, the NRC documentation was used to set the actual guidelines for licensing. They establish the rules that must be followed. The vendor documents and engineering experience helped to set the timeline and cost structure data along with solidifying the actual licensing approach setup.

Once the licensing approach was determined, the cost structure and timeline was developed using Microsoft Project™. To gain the needed data for input into the cost structure and timeline data, the use of engineering experience and vendor documents allowed for what is believed to be the most accurate model of licensing costs and timeline that could be developed. This timeline and cost structure is used to evaluate each of the different scenarios.

Chapter 2 – Literature Review

In this chapter the requirements and common guidelines for the licensing of nuclear reactor fuel are presented. The requirements are spelled out in the Codes of Federal Regulations Section 10 – Energy (10 CFR). These requirements dictate the conditions that need to be met to legally operate a nuclear reactor. From these requirements, guidelines were developed by the NRC to use when evaluating new or current licensing bases against 10 CFR. The guidelines provide acceptable methodology that can be used to minimize the review time associated with changing the licensing bases. The specific subsections that were used are listed below.

There exists more documentation that is not from the NRC that was used to determine the licensing approach and timeline and cost structure. This documentation is listed below and includes the Framatome/ANP "Fuel Qualification Report" and the Westinghouse "WCAP-12610-P-A plus addenda".

1. Current Licensing Requirements

Within Section 10 of the Code of Federal Regulations there are subsections that give the actual requirements for nuclear fuels. These subsections outline the explicit requirements that must be currently met by the nuclear fuels. The current requirements that must be met will be evaluated as if they are the minimum acceptable standard. By setting these as the minimum requirements, it will ensure that any changes made to the fuel, cladding or geometry will be as safe as the current standards.

1.1. 10 CFR Part 50, §50.46

The standard fuel assembly shown in Figure 1 along with the material needs described within 10 CFR 50.46; laid out the configuration requirements. Next, the modeling and testing that is required, along with some of the physical standards that must be met are found in 10 CFR 50.46, and 10 CFR 50 Appendices A and K. There are five main requirements for the current standards. The following were taken from 10 CFR 50.46:

a. The maximum fuel cladding temperature cannot exceed 2200°F.

b. The local cladding oxidation shall not exceed 17% of the total cladding thickness before oxidation. This assumes zirconium is converted to ZrO_2 locally on the cladding wall.
c. The maximum hydrogen generated shall not exceed 1% of the theoretical amount of hydrogen that could be generated during a steam-zirconium reaction in which all of the cladding surrounding the fuel pellets was to react excluding the cladding around the plenum volume.
d. Changes to core geometry shall not affect the ability to cool the core.
e. Long-term cooling of the core will be such that the fuel temperature will be maintained at an acceptably low value and decay heat will be removed for the duration of time required by the long-lived radioactivity remaining in the core.

The above five requirements will now be described in further detail. Number one, the fuel cladding temperature can never reach 2200°F because there will be a loss of mechanical strength and an exothermic, zirconium-steam reaction will accelerate. This steam reaction leads to hydrogen formation and ZrO_2 formation (Equation 1). Hydrogen formation is dangerous because of the potential for explosions and therefore the release of radioactivity. The formation of the zirconium oxide is unacceptable because the cladding is assumed to have no structural strength above 17% oxidization. Because the fuel cladding has no structural strength above 17% oxidation, the fuel cladding could breach and release fuel pellets into the reactor coolant.

$$2H_2O + Zr \longrightarrow ZrO_2 + 2H_2 + heat$$

Equation 1 - Zirconium / Steam Reaction

The second requirement is used to ensure that the formation of zirconium oxide is kept to an acceptable level. The fuel assemblies are only allowed a specific amount of time inside of the reactor. The amount of time is called burnup and it is measured in either Gigawatt days per metric ton of uranium (GWD/MTU) or Megawatt days per metric ton of uranium (MWD/MTU). Fuel assemblies are currently rated for 62 GWD/MTU, which is directly influenced by their localized power level and time spent inside the reactor. At 60-63 GWD/MTU, the fuel cladding is oxidized at close to the 17% threshold.

Number three can be a little misleading. Within a fuel rod, only approximately 80-90% of the volume is filled with fuel. The remaining 10-20% is known as the plenum and is there to moderate pressure buildup of fission gas products within the fuel rod. A standard Westinghouse fuel rod is shown in Figure 3. Therefore to calculate the maximum amount of hydrogen that can be theoretically released, calculate the amount of hydrogen that could be released if all of the cladding were reacted to zirconium dioxide and multiply that by the percentage of the fuel rod that is fueled. In effect this requirement limits maximum fuel damage to 1%.

Number four requires that during a loss of coolant event (LOCA) the core will remain such that coolant and control rods will be able to be injected/inserted into the core. This means that coolant channels and thimble guide tubes will not be grossly distorted and block the flow of coolant or insertion of control rods.

Number five describes the requirement that after an accident, the energy remaining in the core will be able to be dissipated. This is accomplished by circulating coolant to ensure that the fuel does not reach its melting temperature. This is approximately 5400°F for the uranium oxide[2] fuel at its centerline and below 2200°F for the cladding at its surface. These numbers are dependant upon burnup and fuel that has a higher burnup will have the maximum centerline temperature decreased due to lower achievable power density and decay of some fission products.

1.2. 10 CFR Part 100

> This regulation requires analyses to be performed to ensure that during a postulated accident the dosage to those outside the exclusion zone will be within regulatory limits. Specifically, reactors are currently licensed such that no persons outside of the exclusion zone will receive a dose greater than 1500 mREM during a postulated accident.

1.3. 10 CFR Part 50, Appendix A

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[2] NIST Webbook, Phase Change Data
http://webbook.nist.gov/cgi/cbook.cgi?ID=C1344576&Units=SI&Mask=4#Thermo-Phase

The General Design Criteria (GDC) that are specifically applicable to fuel design are GDC 10-13, 20, and 25-28. These GDCs collectively hold that the fuel design criteria remain intact during all normal operations and anticipated operational occurrences. The following criteria are taken directly from 10 CFR Appendix A:[3]

Criterion 10--Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11--Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity. (This is negative feedback on a power transient).

Criterion 12--Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Criterion 13--Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Criterion 20--Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems

[3] Code of Federal Regulations Section 10, Appendix A; Amended December 23, 1999; http://www.nrc.gov

including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 25--Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Criterion 26--Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 27--Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28--Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

1.4. 10 CFR Part 50, Appendix K

Appendix K gives the allowable means to calculate emergency core cooling system (ECCS) needs due to a loss of coolant accident (LOCA). This Appendix lists the applicable methods and equations that are available for use to calculate the ECCS needs during a LOCA without further review by the NRC.

2. Current Licensing Guidelines

2.1. <u>NUREG-0800 – Standard Review Plan Sections 4.2 - 4.4</u>

Sections 4.2 through 4.4 of the Standard Review Plan (SRP) are a guideline for review by the NRC when licensing fuel system design, nuclear design, and thermal and hydraulic design. From these sections the needed documentation and analyses can be ascertained. These documents and analyses ensure that the requirements of the codes of federal regulations are followed when licensing a new fuel, cladding, or geometry.

2.2. <u>Regulatory Guides</u>

Regulatory Guides (RG) are used to give instruction to calculations and analyses of specific areas for nuclear power licensing, i.e. plume models for reactivity release.

The specific regulatory guide is reference when applicable and they will not be discussed in detail here. However, it should be understood that the regulatory guide is NRC approved methodology that is meant as a guideline. These guidelines are meant to allow an easier time of licensing when dealing with common calculations and common problems or questions that arise during licensing.

3. Documentation Supporting Licensing of New Fuel

3.1. <u>"Fuel Qualification Plan", Framatome-ANP/DCS, dated 4/2/2001</u>

3.2. <u>License Amendment Request (LAR) for Catawba and McGuire Nuclear Stations Units 1 and 2, from Duke Power to NRC, dated 2/27/2003</u>

The Fuel Qualification Plan was used to outline Framatome-ANP/DCS efforts gain licensing for MOX fuel. The Fuel Qualification Plan, along with the LAR for Catawba and McGuire, gained licensing approval for four LTAs after two years with the NRC.

4. Documentation Supporting Licensing of New Cladding

4.1. <u>WCAP-12610-P-A plus addenda</u> – Vantage+ Fuel Assembly Reference Core Report and Optimized ZIRLO

4.2. Westinghouse Fuel Rod Design 200 class notes

5. Documentation Supporting Licensing of New Geometries

5.1. <u>WCAP-12610-P-A plus addenda</u> – Vantage+ Fuel Assembly Reference Core Report and Optimized ZIRLO

5.2. Westinghouse Fuel Rod Design 200 class notes

5.3. MIT-NFC-TR-063 – Neutronic Design of PWR Cores with High Performance Annular Fuel; Xu, Zhiwen, et. al., May 2004, MIT/CANES

6. Requirements for Fuel Systems Design

The following are excerpts from the NUREG-0800 Standard Review Plan (SRP), Sections 4.2 to 4.4. The SRP covers "Fuel System Design," "Nuclear Design," and "Thermal and Hydraulic Design." These three sections take their data from the previous "Current Licensing Requirements" section. The SRP is used by the NRC when analyzing changes to the licensing of nuclear power plants. Sections 4.2, 4.3, and 4.4 specifically deal with the nuclear fuel and the design of the core.

6.1. Section 4.2 – Fuel System Design

Fuel system design is divided into four sections. These four sections are "Design Basis", "Descriptions and Design Drawings", "Design Evaluations", and "Testing, Inspection and Surveillance Plans." The first, "Design Basis", is used when determining the limiting values for important parameters so that damage is limited to acceptable levels. The second, "Descriptions and Design Drawings", is used when reviewing fuel systems and places an emphasis on product specifications. The third, "Design Evaluation" is used to evaluate and ensure that "Design Bases" are met during normal operation, anticipated operational occurrences, and postulated accidents. Finally, "Testing, Inspection, and Surveillance Plans", ensures that before, during and after irradiation, all requirements that have been set forth in the previous three areas have been and will continue to be met.

6.2. Section 4.3 – Nuclear Design

The nuclear design is used to develop many of the analyses performed on the core where core performance analyses are concerned. This section is used to confirm the design bases established by the GDC. Specifically the neutronics are important here. From this section the core power distribution, reactivity coefficients, control requirements, rod patterns and reactivity worths, criticality, and pressure vessel irradiation can be determined. Finally the analytical methods used to determine many of the above criteria are addressed.

6.3. Section 4.4 – Thermal and Hydraulic Design

The thermal and hydraulic design section is used to determine the computer calculations that are needed to substantiate reactor analyses. Furthermore, the correlation of experimental data and verification of process and phenomena applied to reactor design are also included. This section is not as in depth as SRP 4.2 but is useful when determining the Thermal and Hydraulic portion of the licensing approach.

Chapter 3 – Licensing Approach

1. Overview

As previously stated in Chapter 1, the standard commercial reactor fuel assembly is made of uranium dioxide fuel; zirconium based cladding; and cylindrical fuel rods held into place by grids. This setup will is referred to as the standard fuel assembly. When a portion of the standard fuel assembly is changed it will be noted. For example, when the new fuel is introduced, it is assumed, that only the fuel is changed, the cladding will remain a zirconium alloy and the geometry will not change. Because of the changes to the standard fuel assembly, the new fuel, cladding, and geometry may require reanalysis of models and commonly used procedures and methodology.

Upon completion of new or updated models, the results will be sent to the NRC. During the time the models are under review by the NRC, there will be a staff of engineers that will answer questions that are posed by the NRC. These questions from the NRC are known as requests for additional information (RAIs) and need to be answered in a timely fashion because NRC review time is a critical path task and must be completed before further phases of the licensing approach can be completed.

The word "licensee" is used to denote the holder of the operating license for a commercial nuclear plant. The licensee is the entity that can apply for changes to their license and who will procure the new fuel assemblies for use in their reactor.

1.1. Advantages of Increased Performance from New Fuel, Cladding, or Geometry over the Standard Fuel Assembly

Before a new fuel, cladding, or geometry is even considered, its advantages over the standard fuel assembly will need to be determined. Preliminary calculations of the cost/benefit of the changes and manufacturing needs should have been completed. This will ensure that before changes to existing models, etc are started, any logistical or obvious errors are resolved. To aid in the analyses, physical properties of the new fuel, cladding, or geometry should be known. Irradiation data may or may not be addressed in this initial phase. The advantages for new fuel, cladding or geometry will be addressed individually. The most common reasons for change are listed below.

1.1.1. **New Fuel Goals and Advantages**

There are many possible advantages to selecting a new fuel other than uranium dioxide. Some of the advantages for a new fuel include increased density, increased thermal conductivity, decreased specific heat, and increased fuel melting temperatures.

Increased fuel density allows for a higher heavy metal loading without increasing the enrichment. Because of the higher heavy metal content, the new fuel can be used for higher burnups at the same enrichment level. The benefit of this is that longer fuel cycles are available to the power plant. Fuel cycle lengths above the current maximums of 20-24 months with less fuel needing to be reloaded for each cycle would be the goal. This not only reduces fuel costs due to lower enrichments, but increases power plant availability by reducing the number of refueling outages.

Increased thermal conductivity fuel results in lower fuel centerline temperatures than would be expected with UO_2. These lower fuel temperatures decrease the fission gases released from the fuel. This is especially important if the new fuel has a higher density because the higher heavy metal loading would increase the fission gasses that would be produced. The decrease of the fuel centerline temperature would also help somewhat during a loss of coolant accidents (LOCA) due to lower residual heat content. This lowers the initial heat transfer needs important during a LOCA.

The decreased specific heat also allows for a lower stored energy reserve for the fuel. This coupled with the increased thermal conductivity allows for a significant LOCA improvement over the standard fuel. This LOCA improvement allows for the fuel to be safer at beginning-of-life when the energy stored in the fuel is higher due to higher power per foot operation.

Increased fuel melting temperature, and a higher allowable fuel centerline temperature provide a larger safety margin during a LOCA. The increased margin of safety during a LOCA is very important. This allows for more power within the core without sacrificing the safety margin for fuel melting. Therefore, if the fuel were to run hotter, increases to the mass flow of coolant through the core allows the current temperature and pressure of a standard PWR, 650°F and 2200 psi, to remain unchanged, while extracting more power from the same size core.

1.1.2. **New Cladding Goals and Advantages**

A new cladding can be viable if it increases the amount of time the fuel assembly can remain in the reactor, has better heat transfer properties, lessens the possibility of hydrogen formation, has increased resistance to oxide formation, has a higher operating point or has better economics for the vendor and the licensee. If one of these can be met and the cladding at least meets the current requirements of the NRC, then the new cladding would be considered for licensing.

Currently the maximum amount of time that a fuel assembly can remain in the reactor is 62 GWD/MTU. The reason that this amount of time has been chosen is because the current cladding becomes too "worn out" after this long irradiation period. The cladding becomes embrittled due to increased ZrH_2 within the cladding. This hydride formation decreases the strength, which increases the possibility that the cladding could fail if there is an accident. Once the cladding is removed from the reactor it still needs to ensure that it will not fail while in the spent fuel pool and later when moved to long term storage either in dry cask storage or in a geologic repository.

Because of the 62 GWD/MTU maximum irradiation limit, commercial power reactors operate on an 18 month refueling cycle. If the cladding could withstand higher irradiation, then it would be possible for the licensee to increase their refueling outages to once every 24 months. This would be valuable to the licensee since they would have to refuel only 20 times over the course of their 40 year lifetime as opposed to approximately 28 refuelings for an 18 month cycle. At a cost of approximately $30 million for each 30 day outage, a 24 month cycle could save a licensee $240 million over the lifetime of the power plant.

The increase in the length of time that fuel is allowed in the reactor also decreases the amount of spent fuel generated. This decrease in the amount of spent fuel provides a significant economic benefit to the DOE which would ultimately dispose of the spent fuel.

An increase in the heat transfer properties of the cladding would lead to an increase in the safety margin for the core. If the thermal conductivity of the cladding were to increase, the centerline temperature of the fuel would decrease somewhat and therefore the stored energy in the core for large break LOCA calculations would be decreased.

A cladding that eliminates the formation of hydrogen production would end the possibility that the there could be a hydrogen explosion. This would eliminate the requirement that the fuel cladding not reach 2200°F as long as the cladding will not undergo a reaction with water to form hydrogen. (See Equation 1 - Zirconium / Steam Reaction)

A decrease in the oxide formation of the cladding would allow for higher mechanical strength. This is especially important for higher burnup fuel during accident conditions. This is because one of the current limitation to higher burnups is the >17% oxide layer that will form and therefore make the fuel rod unable to withstand an accident at the end-of-life. If the amount of oxide formation were limited or eliminated the fuel would be safer at higher burnups.

An increase in the cladding operating temperature would allow for the fuel cladding to reach higher temperatures during accident or off-normal operating conditions. This increase in temperature would give a higher safety margin if a LOCA or departure from nucleate boiling (DNB) were to occur, since the cladding would not deform as quickly and would result in less potential for fuel damage and allow for a longer amount of time before safety injection would need to occur.

1.1.3. **New Geometry Advantages**

There are advantages that must be considered when changing the geometry of the fuel rod or fuel assembly over those currently licensed. These include increasing the heat transfer area for fuel rods, decreasing the pressure drop through the core, and increasing the mixing through out the fuel assembly.

By increasing the heat transfer area of the fuel rod, greater energy per volume can be extracted than could be extracted from a core that uses cylindrical fuel. For example, if the fuel were changed from a cylinder to an annulus, the heat transfer area per volume would be dramatically increased. This would allow for increased thermal output from the core with out increasing the chance for DNBs.

In order to allow for greater mixing of the coolant within the fuel assembly, the standard fuel assembly has inter-spaced mixing grids. The problem with this arrangement is that it leads to large pressure drops through the core at high coolant flow rates while increasing heat transfer only at the locations of the grids. Therefore, if the geometry of the fuel assembly were manipulated to allow for a lower pressure drop and increased flow mixing, higher coolant flow rates and higher fuel rod heat duties would be allowed, thus increasing the power from the core.

1.2. Recent Deviations from Current Licensing Requirements

1.2.1. **New Fuel Deviations**

There are currently two deviations from the standard UO_2 fuel used in commercial reactors. The first was the use of a ThO_2/UO_2 mixed oxide core in Indian Point in the early 1960's. The second is the Department of Energy (DOE), along with Framatome-ANP/DCS (Duke, COGEMA, Stone & Webster), licensing of a mixed oxide (MOX) fuel. MOX fuel, in this situation, is a combination of weapons grade (WG) plutonium oxide (PuO_2) and depleted or natural UO_2. March 3, 2005 Duke Energy was granted permission to use four lead test assemblies (LTA) of MOX fuel in their Catawba Nuclear Station.

1.2.2. **New Cladding Deviations**

There have been a few deviations from the current requirements; however, except for the change from stainless steel clad to zirconium clad in the early 1960s, these deviations have been a zirconium based alloy. No later deviation from the zirconium alloys has been found. Currently the use of M5^{TM} cladding (zirconium based) is being phased in by Framatome. Some of the needs for a new cladding will be taken from the submittals to the NRC for M5^{TM} cladding. Also, archival Westinghouse WCAPs for ZIRLO^{TM} will also be used.

1.2.3. **New Geometry Deviations**

No deviation from the standard fuel assembly has been found for light water reactors (LWRs). The DOE has looked at the next generation of reactors using spherical fuel, which would be used to fuel any pebble-bed technology reactor; however these are used for gas-cooled reactors. Therefore, the idea of new fuel geometry for LWRs has not been explored. The only changes made to the fuel assembly are the addition of inter-fuel mixing vanes.

1.3. Supporting Documentation

A listing of all supporting documentation for the new fuel, cladding, and geometry can be found in Chapter 2 – Literature Review.

2. Outline for the Licensing of New Fuel, Cladding, and Geometry

The next step is to determine a methodology to license changes to the fuel, cladding and geometry. Before this is done there are several steps that will have already been completed. These include an economic evaluation to determine the advantages over the currently licensed setup, determining related licensing deviations, and gathering the needed supporting documentation.

The licensing of these changes to the standard fuel assembly will be divided into four separate phases. Each phase will provide a base for the latter phases to follow. The first three phases will each build upon the previous phases until the ultimate goal of a full core loading is obtained. Phase four will attempt to stretch the boundaries of current burnup limits in an effort to increase the economic advantage of the new fuel, cladding, geometry. These phases will follow the guidelines laid out in NUREG-0800 Standard Review Plan, Section 4.2 – 4.4. From the guidelines a comprehensive list will be established that can be used as a tool to complete the licensing requirements needed to fulfill the licensing requirements.

2.1. Development and Testing – Phase 1:

The tasks in Phase 1 are:

1. Development of a New Fuel, Cladding, or Geometry Specification
2. Analysis of a Standard Fuel Assembly with the New Fuel, Cladding, or Geometry
3. Core Performance and Safety Evaluations
4. Initial Confirmation through Lead Test Assemblies with Test Rods in a Research Reactor

Phase 1 meets the initial needs for licensing that will be the support for the rest of the licensing effort. It is the development of specifications, modification of analyses and models, and evaluations and testing that will lead up to and through the placement of Lead Test Assemblies with Test Rods (LTATR). The LTATRs will require post-irradiation examination (PIEs) to determine how the rod behaved during the fuel cycle. The PIEs will require poolside and hotcell examinations. The PIEs for the LTATR will allow for partial confirmation of the updated tools and analyses and will provide sufficient operating experience to allow for the commencement of Phase 2.

For the new cladding or fuel, the installment of small coupons or test rodlets of the new cladding or fuel material within a reactor is advisable. This would be done prior to the implementation of the LTATRs. These coupons or fuel pellets would be irradiated and then sent for analysis to determine their post-irradiation physical properties.

2.2. LTA Loading – Phase 2:

Confirmation through Lead Test Assemblies in a Commercial Reactor

Phase 2 will involve the licensing of the Lead Test Assemblies (LTAs). Data gathered from the LTAs will allow for further confirmation of the models and analyses. Again, much of the data gathered will be in the form of PIEs performed to examine the fuel assembly and enclosed fuel rods. The PIE will require poolside and additional hotcell examinations. Several LTA programs in various commercial plants will be necessary to confirm applicability of the fuel to various plant conditions and fuel management conditions, however, hotcell examinations should not be required unless specifically requested by the NRC. The examination and subsequent confirmation of models and analyses will then allow for the commencement of Phase 3.

2.3. Full Core Loading – Phase 3:

Application and acceptance for Licensing with Full Core Loading

Phase 3 is the completion of the requirements needed for full core loading of the new fuel, cladding or geometries. This phase is the preparation, submittal and approval steps needed to license the new fuel. The completion of this stage is the approval for use in either BWRs or PWRs up to the maximum allowable fuel burnup. (Currently 62 GWD/MTU)

2.4. Increased Burnup and Synergistic Effects– Phase 4

Application and acceptance for higher burnups

Phase 4 is the final milestone. This milestone allows for full advantage of the new fuel, cladding or geometry to be realized. For instance, the fuel and cladding could be run beyond the allowable 62 GWD/MTU up to 100 GWD/MTU allowing for a 24 month fuel cycle. This step takes the fuel, cladding, or geometry changes to a fully optimized product for use in commercial power reactors.

3. Development and Testing – Phase 1

3.1. Development of New Fuel, Cladding, or Geometry Specification

The fuel, cladding, or geometry specification is the base for the entire licensing effort. Many of the analyses and coding that are to be performed rely on an accurate fuel performance model and material properties. Each of the three changes to the standard fuel assembly will require varying degrees of changes to the current models.

3.1.1. Description of the fuel rod

The first step is to develop a description of the fuel rod. An acceptable fuel design description is given when at least the following has been met. (From SRP 4.2 Section II.B)

i. Type and metallurgical state of the cladding [C]
ii. Cladding outside diameter[C,G]
iii. Cladding inside diameter[C,G]
iv. Cladding inside roughness[C]
v. Pellet outside diameter[C,G]
vi. Pellet roughness[F]
vii. Pellet density[F]
viii. Pellet resintering data[F]
ix. Pellet length[F,G]
x. Pellet dish dimensions[F]
xi. Burnable poison content[F]
xii. Insulator pellet parameters
xiii. Fuel column length[F,G]
xiv. Overall rod length
xv. Rod internal void volume[F]
xvi. Fill gas type and pressure[F]

xvii. Sorbed gas composition and content[F]

xviii. Spring and plug dimensions[F,G]

xix. Fissile enrichment[F,G]

xx. Equivalent hydraulic diameter[G]

xxi. Coolant pressure

xxii. Design Specific Burnup Limit[F,C]

Possible changes to current descriptions – ‘F’ fuel; ‘C’ cladding; ‘G’ geometry

Section II.B also requests the following drawings be submitted in order to facilitate review. These are typically illustrative type drawings that would be found in a commercial power plant’s Final Safety Analysis Report (FSAR).

i. Fuel assembly cross section

ii. Fuel assembly outline

iii. Fuel rod schematic

iv. Spacer grid cross section

v. Guide tube and nozzle joint

vi. Control rod assembly cross section

vii. Control rod assembly outline

viii. Control rod schematic

ix. Burnable poison rod assembly cross section

x. Burnable poison rod assembly outline

xi. Burnable poison rod schematic

xii. Orifice and source assembly outline

Specifically for changes to the fuel and cladding, the size and shape of the fuel rods and fuel assembly are not expected to change, at least initially. For geometry changes, there are likely to be more parameters to consider. For example, if a change in the overall shape of the fuel rod is made, there may be a center water channel inside or fins on the outside whose dimensions would need to be added to the list that was taken from SRP 4.2 section II.B.

3.1.2. Comparison of the new fuel, cladding and geometry to UO_2, zirconium alloys, and the standard fuel assembly geometry

A comprehensive evaluation of both the advantages and disadvantages compared to the standard fuel assembly will be completed. This section supports the decision as to whether a specific fuel, cladding or geometry fuel will be viable when subjected to the harsh environment when irradiated inside of a LWR. Previously, only the advantages were considered. Here a list of disadvantages should also be looked at. These disadvantages should not preclude the licensing of the new fuel, but should be evaluated because of their probable questioning by the NRC. For example, an increase in density is advantageous, but the resulting probable increase in possible dosage to those outside the exclusion area is disadvantageous. The result is that the exclusion area should be increased, but this alone should not eliminate the new fuel from being used.

3.1.2.1. Analysis of a new fuel against uranium dioxide

Table 1 below contains the physical properties of standard UO_2 fuel.[4] From this an evaluation of the physical properties of the standard UO_2 fuel against the new fuel will be done. The pros and cons of either increasing or decreasing the specific physical properties will be evaluated.

[4] STD-CP-03-30 – [Proprietary], Task, K.D., November 13, 2003, Westinghouse Science and Technology Division

Table 1 - Physical Properties of UO_2

	Uranium Dioxide (UO_2)
Density (g/cm^3)	10.954 (@25°C)
Melting Point (K)	2673-3173
Specific Heat (J/mol·K)	63.87 (@25°C)
Thermal Conductivity (W/m·K) (Low Temp.)	9.76 (@27°C)
Thermal Conductivity (W/m·K) (High Temp)	3.144 (@1027°C)
Nuclear Cross Section (mb)	0.19 (^{16}O)

One of the first physical properties that should be considered when analyzing new fuels against UO_2 is the change in density. A change in density directly translates to an increase or decrease in heavy metal loading. This change in heavy metal loading affects the amount of enrichment needed for the new fuel rods to have the same energy level as that of equivalent volume UO_2 rods. The definition of heavy metal loading is the amount of uranium or plutonium that is loaded into the core. When there is a higher density fuel, a lower U-235 enrichment can be used to have the same amount of fissionable material inside of the core. Likewise, a lower density fuel would need a higher enrichment of U-235 for the same energy level.

The enrichments costs directly affect the cost of the fuel. The cost of enrichment is not linear and increases exponentially with higher enrichments. Currently, the highest enrichment that can be used for commercial fuel is 5% U-235. Therefore, with uranium dioxide fuels, the highest density of fissionable material that can be achieved is 0.547 g/cm^3. Comparably, the highest density uranium fuel that could theoretically be used is pure uranium. Pure uranium has a density of 18.9 g/cm^3 and would lead to 0.945g/cm^3 fissionable loading which translates to a 73% increase over that currently available with UO_2 fuel. This increase in fissionable material without exceeding the 5% enrichment limit mandated by the NRC, allows for the higher density fuel rods to either have a higher power with the same burnup or to have the same power but to reach higher burnup levels. Comparatively, a lower density fuel would either have to sacrifice burnup for power level, or would need to increase the enrichment.

If the thermal conductivity of a lower density fuel rod were higher, then it could have the same power as an equally enriched uranium dioxide fuel rod. This would especially be true if the new fuel were to have its thermal conductivity increase as the fuel temperature increased, since UO_2 thermal conductivity decreases as the fuel temperature increases. Note, however, that the fuel life would be reduced due to the overall decrease in U-235 loading.

One concern with increasing the heavy metal loading is the increased amount of radioactive U-235 that is placed into the core. This will affect the 10 CFR 100 calculations that govern the exclusion area that must be kept around nuclear power plants. Also, with increased density, the LOCA scenario must be reevaluated since the higher U-235 content in the core could lead to less than adequate emergency core cooling system (ECCS) capability. This is because part of the ECCS is to remove the decay heat from the radioactive isotopes in the core. Since there could now be more U-235 in the core (assuming that a higher density fuel rod is enriched to 5%), the needs of the ECCS for decay heat will be increased, specifically for any beginning-of-life accidents when the fuel will have much higher U-235 loading. At the end-of-life, if the higher density fuel is at higher burnup, the amount of U-235 within the fuel should be approximately similar.

The next property that must be evaluated is the specific heat. The specific heat is important because LOCA calculations depend on the amount of stored energy in the core. Therefore, new fuels with lower specific heats would require slightly less ECCS capability during a LOCA than UO_2. Continuing on the thermal properties, the thermal conductivity is important when calculating maximum fuel centerline temperatures. Unfortunately, the thermal conductivity of UO_2 decreases as the temperature increases. This leads to a scenario that is especially worrisome. During a LOCA when the fuel is heating up; the ability for the fuel to shed its heat is decreased because of the lowered thermal conductivity. Therefore, any fuel that would have the thermal conductivity increase when the temperature increased would be particularly valued. Finally, a fuel with a lower specific heat and increasing thermal conductivity would lead to a larger margin of safety in the large-break loss of coolant accident (LBLOCA).

Finally, the nuclear cross section of the element or molecule that is bonded to the uranium needs to be considered. The nuclear cross section of the partner is important because high nuclear cross sections lead to increased parasitic neutron loss. This is especially important when delayed neutrons are needed, like those in a light water reactor. Therefore a decrease in the nuclear cross section would help to minimize parasitic neutron loss within the fuel and would increase the fuel life with the same enrichment.

3.1.3. Analysis of new cladding against current zirconium alloys (ZIRLO)

Error! Reference source not found. below contains the physical properties of standard ZIRLO cladding against which a new cladding can be compared.

Table 2 - Physical Properties of ZIRLO cladding

	Optimized ZIRLO
Density (g/cm^3)	6.562
Operating Temp (°C)	<1200°C
Uncontrolled Metal-Water Reaction	Yes (above 1200°C) Exothermic Produces H_2 gas
Specific Heat (J/mol·K)	~300 (@20°C) ~750 (@870°C) (Max)
Thermal Conductivity (W/m·K)	Irradiated 15 (@20°C) 34 (@1200°C)
Thermal Expansion Coefficient	$5.75(10^{-6}C^{-1})$
Nuclear Cross Section (mb)	194 (Natural Zr)

If the density increases, the mass of the fuel assembly could increase, and if the cladding is still a zirconium based alloy, the mass of zirconium in the cladding could increase. The opposite scenarios would occur if the density were to decrease. First, an increase in the mass of the fuel assembly could cause fuel handling problems for the licensee. In an extreme case, upgrading the fuel handling cranes could be necessary to accommodate the increased weight. The increase in mass of zirconium or most metals in the core would cause other problems such as an increase in hydrogen production during an accident or an increase in parasitic neutron losses. Based on these considerations, it is unlikely that an increase in density would be a favorable trait for a new cladding.

The use of a cladding with a higher operating temperature is a favorable option. The current operating temperature maximum is 1200°C for zirconium claddings. The reason is that the exothermic self-sustaining metal-steam reaction that produces H_2 accelerates above 1200°C. Therefore an alloy that does not undergo the zirconium-steam reaction would be very valuable. A decrease to the operating temperature cuts into the operating safety margin of the core. However, if this decrease was small, but the possibility of a zirconium-steam reaction was eliminated, this cladding would have a good possibility of being licensed.

The best cladding that could be developed here would be one that vastly increased the operating temperature and eliminated the metal-steam reaction. This cladding would ideally have the ability to withstand a LOCA and allow the fuel to cool without melting or shattering upon emergency core cooling system water injection.

The coefficients of thermal expansion for current and new cladding need to be evaluated. If they are essentially equal, it will allow for the use of the current cladding alloy as thimble and guide tubes for initial testing of the new cladding within the standard fuel assembly. This minimizes the changes that will need to be made. If they are wildly different, then more engineering analysis will have to be performed to evaluate where changes can be made to allow for testing.

Finally, a change to the nuclear cross section of the cladding will affect neutronics. An increase to the nuclear cross section of the cladding could lead to an increase in parasitic neutron loss. This not only decreases the chemical shim or moderation needed inside the core, but also negatively affects the economics of the fuel by leading to a shorter fuel life. A decrease in neutron cross section would decrease the parasitic neutron losses and would increase the chemical shim needed to control the core. However, a lower cross-section also leads to longer fuel life and therefore better fuel economics.

3.1.4. Analysis of new geometry against the current gridded fuel assembly with mixing vanes.

There are geometry changes to the standard gridded fuel assembly and cylindrical fuel rod that would increase the heat transfer area and decrease the pressure drop through the core.

Changes to the available heat transfer area can have varying effects. First, by increasing the heat transfer area, more power can be extracted from the same size core. However, an increase to the mass flow rate of coolant must occur. In order to do this in currently operating plants, there is a distinct possibility that the reactor coolant pumps would need to be upgraded. Also, an increased amount of power would force the secondary system to also become larger. This would probably cause one of two scenarios. Either a new, larger secondary system having larger steam generators and turbine-generator system would be installed, or a second balance of plant would be build adjacent to the current structure that would be able to convert the excess energy into electricity[5]. Another problem for current plants would be the increased pressure drop through the core. Because not only the area, but the mass flow rate of coolant is increased, a much larger pressure drop is inevitable. This could be avoided by changing other design aspects of the fuel geometry to allow for greater mixing with a lower pressure drop. Because of the changes to the heat transfer properties there will most likely need to be an increase of the flow rate through the core along with an increase in the ΔT. Therefore, the largest obstacle to the licensing of a new geometry that increased the heat transfer area would be the larger core pressure drop resulting from an increased flow. For new plants, the higher heat transfer rate per volume will allow more power to be produced from a smaller primary system and increased power output to the secondary system. This lowers the capital cost of new plants.

Decreasing the heat transfer area in the core would decrease the pressure drop through the core. To accomplish this, a smaller surface area to mass ratio fuel rod would be used. This would allow for more mass of uranium to be in the core than was previously available which would allow a lower enrichment to be used. For a decreased heat transfer area fuel assembly to be viable, the lowered uranium cost would need to offset the decreased heat transfer area and power output.

Fuel geometries that increase mixing in the core while keeping the pressure drop and flow constant would allow for higher heat transfer coefficients

[5] Patent Disclosure: U.S. Patent No. 6,909,765 Lahoda, Edward J., Westinghouse Electric Company LLC, Pittsburgh, PA; June 21st 2005

and more power output. The result would be an increase in the temperature of the coolant. The higher operating temperature increases the thermodynamic efficiency of the plant.

Finally, the neutronics of changing the heat transfer area needs to be evaluated. If heat transfer is increased by adding cladding mass, then there will be an increased amount of parasitic neutron absorbers. If the heat transfer is changed by reconfiguring the geometry of the rod, i.e., an annular rod, an increase of coolant and possibly cladding must be considered. Both of these will changes will result in slightly worse neutronics in comparison to the standard fuel assembly. The decrease in neutronics is because of a larger amount of parasitic neutron sources now within the fuel assembly.

3.2. Analysis of a Standard Fuel Assembly with the New Fuel, Cladding, or Geometry

Now that the advantages and disadvantages of the new fuel, cladding and geometry have been established, an analysis of the models and analyses that must be performed will be determined. These models and analyses will provide the needed data that will allow for the licensing application for the Lead Test Assembly with Test Rods (LTATR). These models will be updated with the data from the test reactor.

The new fuel and cladding will have different operational properties that must be accounted for in the standard fuel assembly. This will require new fuel and cladding, models and analyses. Once the new models have been completed, they will need to be resubmitted to the NRC for evaluation and acceptance. One of the issues involved in doing this will be the requirement to have models that are already approved to have their source codes opened to NRC scrutiny again.

The scale of the physical changes varies with the proposed changes. For fuel, changes the rod length may need to be changed to accommodate increased fission gas release, the increased fuel assembly weight accounted for, etc. Specific changes will be addressed by the fuel design engineers. For cladding, there are no major changes other than providing new properties.

For changes in geometry, there will be large scale changes due to the complex nature of flow patterns for heating, cooling and vibration. In a worst case scenario, full scale reflood tests would need to be performed before an LTA could be developed. Even if a reflood test is not needed by the NRC, significant changes to the modeling and analyses will still be needed. Because of this, there will be greater costs and time needed for the preliminary modeling and analysis of the geometry changes.

The software programs that are needed to model the new fuel, cladding and geometry are divided into four categories. These categories are fuel performance software, core physics software, LOCA and non-LOCA evaluation models, and thermal-hydraulic/mechanical analysis.

3.2.1. Fuel Performance Software

Fuel performance software is used to determine the power histories for burnup cycles, rod internal pressures; essentially all of the parameters needed when determining if the fuel will behave as expected. For example, the current fuel performance software used by Westinghouse is PAD 4.0 and it is written for UO_2 fuel within a Zr-based cladding. Other vendors will have similar fuel performance codes.

Parameter changes include thermal conductivity, melting point, radial power profiles and fission gas release. For the new cladding, new physical properties, including thermal conductivity, steady state stress and strain, and DNB correlations will all have to be included.

If the geometries are radically changed, it is expected that there will need to be a rewriting of the code. In addition, tests to verify the new code will be required. This will increase the cost of the geometry licensing beyond that of the fuel and cladding licensing.

3.2.2. Core Physics Software

The current core physics software used by Westinghouse to predict Beginning of Cycle (BOC) Hot Zero Power Physics Tests and Hot Full Power calculations is Phoenix-P/ANC for PWRs and PHOENIX/POLCA for BWRs. These tests provide control bank worth, isothermal temperature coefficients, critical boron concentrations, and core power distributions. Control bank worth is an analysis that provides the licensee with the moderating effects for each bank of control rods. Isothermal temperature coefficients are used to determine the reactivity of the core at specific temperatures. Critical boron concentration is used to determine the amount of soluble boron that should be added to the core to eliminate excess reactivity. Core power distributions are the actual expected power of the core during operation and can be calculated for each refueling cycle. From these calculations the fuel engineers determines the core loading patterns. These become especially important when determining where to load the new fuel.

For the new fuel, software to predict many of the three-dimensional neutronics associated with commercial power reactors will also be needed. This software is the basis of many of the computer codes used in the analyses of reactor physics, design and control.

For the new cladding, the software will probably not change other than to enter the new physical properties of the cladding. If this cladding is zirconium based, the changes to the software should be much less than that for a new clad material.

The new geometry could be problematic. If a change is made to the shape of the fuel rod, the entire model could be invalidated. This would cause a complete rewriting for the new geometry. However, this has been done before by students at MIT in order to prove some of their theories on the effects of changes to geometry without too much difficulty. The difficulty involved in getting the NRC to approve these new models, however, has not been determined.

3.2.3. LOCA Evaluation Model

For LOCA evaluations, two scenarios will be evaluated. These are the small break LOCA or SBLOCA and the large break LOCA or LBLOCA. For the LBLOCA, the specific heat and thermal conductivity of the fuel is most important. During a SBLOCA, the total remaining reactivity in the core that must be accounted for is most important.

If the new fuel causes an increase in the mass of U-235 in the core, then the SBLOCA will have to be remodeled. However, if initial testing is done where the total U-235 in the core is kept the same as in a standard fuel assembly this is not really a problem. The LBLOCA will have to be reevaluated because it is almost certain that any new nuclear fuel will have different thermal conductivity and specific heat.

For new cladding, this section is only important if the cladding will more readily undergo a steam reaction than zirconium. If this is not the case, and if the operating temperature were increased, then it is possible that all LOCA evaluations would be easily redone for a new cladding. If the steam reaction proceeds more rapidly tan it does with zirconium, then new models backed by extensive test data will be required.

For a new geometry, the LOCA analysis becomes more challenging. For instance, if the new geometry includes an annular water channel, a LOCA could lead to superheating of the coolant within the annular channel. This would lead to steam which could lead to a hydrogen gas release.

3.2.4. Non-LOCA Evaluation Model

The applicable models for non-LOCA evaluation will be evaluated at a later time. They are not important for allowing an LTATR in a test reactor, or for an LTA in a power reactor. Therefore, non-LOCA evaluation models will be evaluated during the same time that other analyses are evaluated and will not add to the critical path time since it may not be needed for upwards of 8-10 years.

3.2.5. Thermal-Hydraulic / Mechanical Analysis

Thermal-hydraulic and mechanical analyses are performed to ensure that the fluid flow through and stress placed upon the core are within the guidelines of the licensee's FSAR. Therefore, these analyses will be changed most by a reconfiguring of the fuel assembly.

For a new fuel, there are no changes expected to the exterior of the standard fuel assembly since only changes are needed to the interior of the fuel rod. However, if the power density is increased within the fuel rod, analysis of new DNB correlations will be required for licensing. Also, an increase or decrease to the fuel density will affect the fuel assembly lift off analysis that is performed to ensure that the upward force of the coolant moving through the core does not lift the fuel assemblies off of the bottom of the reactor vessel internals. If the new fuel has an increased density, lift-off will not be a problem. However, a decreased density fuel will need to be reanalyzed for fuel assembly lift-off.

For a new cladding, the physical properties of the cladding will have to be evaluated. Since the fuel rod and assembly are not being changed, it is expected that changes to the thermal conductivity, heat capacity, and the fluid friction exerted by the rod will be the most significant. Because of these changes it is important that the DNB correlations and condition III and IV transients be reevaluated. Specifically, the ΔP vs. heat transfer correlations will need to be developed.

It is expected that all codes will need to be rewritten for a changed geometry. In order to keep the timeline short, the cost of redoing models for the new geometry will be increased due to greater man power required. This will be discussed in the Timeline and Cost Structure section.

3.3. Core Performance and Safety Analysis

The models that were developed in the previous section for changes within a standard fuel assembly envelope will now feed into the analyses performed in this section. These analyses are what the licensee will use during normal operations, anticipated operational occurrences, and accident conditions. Most of these analyses are already built into their respective models and should only require small amounts of rewriting.

3.3.1. Evaluation of Fuel and Core Design

3.3.1.1. Core Design

This section was developed from the requirements laid out by the SRP 4.2. Each of the subsections below is taken directly from the guidelines out of the SRP. The subsections are the analyses that need to be performed when doing core design analyses.

3.3.1.1.1. Power Distribution
- 3.3.1.1.1.1. Normal, Steady State, Load-Follow Transients
- 3.3.1.1.1.2. Axial, Radial, and Local Distributions
- 3.3.1.1.1.3. Peaking Factors – Transient / Accident
- 3.3.1.1.1.4. Effect of Fuel Densification

3.3.1.1.2. Control Rod Patterns and Reactivity Worths
- 3.3.1.1.2.1. Control Patterns
- 3.3.1.1.2.2. Allowable Deviations
- 3.3.1.1.2.3. Maximum Worths
  - 3.3.1.1.2.3.1. Individual Rod
  - 3.3.1.1.2.3.2. Rod Banks

3.3.1.1.3. Fuel Assembly Criticality
- 3.3.1.1.3.1. K_{eff} - Determine for both dry and immersed in water
  - 3.3.1.1.3.1.1. Single Assemblies
  - 3.3.1.1.3.1.2. Groups of Adjacent Assemblies

3.3.1.1.4. Reactivity Coefficients

- 3.3.1.1.4.1. Moderator Coefficient
- 3.3.1.1.4.2. Doppler Coefficient
- 3.3.1.1.4.3. Power Coefficient

3.3.1.1.5. Pressure Vessel Irradiation

- 3.3.1.1.5.1. Neutron Flux Spectrum above 1MeV
  - 3.3.1.1.5.1.1. Inside Core
  - 3.3.1.1.5.1.2. Core Boundaries
  - 3.3.1.1.5.1.3. Inside Pressure Vessel Wall
- 3.3.1.1.5.2. Assumptions used in the Calculation
  - 3.3.1.1.5.2.1. Power Level
  - 3.3.1.1.5.2.2. Use Factor
  - 3.3.1.1.5.2.3. Fuel Type – Complete Description of the Fuel
  - 3.3.1.1.5.2.4. Design Life of Vessel
  - 3.3.1.1.5.2.5. Geometric Modeling
    - 3.3.1.1.5.2.5.1. Reactor
    - 3.3.1.1.5.2.5.2. Support Barrel
    - 3.3.1.1.5.2.5.3. Water Annulus
    - 3.3.1.1.5.2.5.4. Pressure Vessel

3.3.1.1.6. Decay Heat

3.3.1.1.7. Reactivity Control and Compensation

- 3.3.1.1.7.1. Long-term changes
- 3.3.1.1.7.2. Hot Zero Power to Cold Shutdown
- 3.3.1.1.7.3. Full Power to Zero Power
- 3.3.1.1.7.4. Xenon Production
- 3.3.1.1.7.5. Soluble Boron Concentration
- 3.3.1.1.7.6. Burnable Poison

3.3.1.2. Fuel Rod Design

This section was developed from the requirements laid out by the SRP 4.3. Each of the subsections below is taken directly from the guidelines out of the SRP. The subsections are the analyses that need to be performed when doing fuel rod design analyses.

- 3.3.1.2.1. Fuel Performance Analysis
  - 3.3.1.2.1.1. Fuel Temperatures
  - 3.3.1.2.1.2. Densification Effects
  - 3.3.1.2.1.3. Fuel Rod Bowing
  - 3.3.1.2.1.4. Structural Deformation
  - 3.3.1.2.1.5. Rupture and Flow Blockage
  - 3.3.1.2.1.6. Fuel Rod Pressure
  - 3.3.1.2.1.7. Metal/Water Reaction Rate
  - 3.3.1.2.1.8. Fission Product Inventory

- 3.3.1.2.2. Fuel Rod Analysis
  - 3.3.1.2.2.1. Cladding Stress
  - 3.3.1.2.2.2. Cladding Fatigue Life
  - 3.3.1.2.2.3. Cladding Creep Collapse
  - 3.3.1.2.2.4. Cladding Transient Strain
  - 3.3.1.2.2.5. Cladding Oxide Thickness
  - 3.3.1.2.2.6. End-of-Life Fuel Pin Pressure
  - 3.3.1.2.2.7. Cladding Hydriding
  - 3.3.1.2.2.8. Prevention of Corrosion Products
  - 3.3.1.2.2.9. Fuel Rod Growth
  - 3.3.1.2.2.10. Fretting
  - 3.3.1.2.2.11. Cladding Overheating
  - 3.3.1.2.2.12. Fuel Pellet Overheating
  - 3.3.1.2.2.13. Pellet/Cladding Interaction (PCI)

3.3.1.2.2.14. Cladding Melting

3.3.1.3.Thermal-Hydraulic Analysis

This section was developed from the requirements laid out by the SRP 4.4. Each of the subsections below is taken directly from the guidelines out of the SRP. The subsections are the analyses that need to be performed when doing thermal-hydraulic analyses.

3.3.1.3.1. Critical Heat Flux Correlation

3.3.1.3.2. Departure from Nucleate Boiling Ratio

3.3.1.3.3. Hydraulic Compatibility

3.3.1.3.4. Core Pressure Drop

3.3.1.3.5. Fuel Assembly Lift

3.3.1.3.6. Cross Flow Velocities

3.3.1.4.Mechanical Analysis

Mechanical analyses of the fuel rod and assembly will have to be performed. These analyses will complete the following sections "Fuel Rod Analysis" and "Fuel Assembly Analysis." Many of the needs are already analyzed for fuel assemblies that have zirconium-based cladding. The remaining testing will need to be performed inside a test reactor. The specific licensing requirements for the test reactor are advisable by the holder of the test reactor license and need to be determined on a case by case basis. Irradiation in a test reactor for the new geometry is very important but will need to again be approved by the test reactor licensee. Important to this section is the need for accurate modeling in the thermal-hydraulic/mechanical section.

3.3.1.4.1. Fuel Rod Analysis

3.3.1.4.1.1. Fuel rod axial growth and shoulder gap closure

3.3.1.4.1.2. Fuel rod shipping and handling

3.3.1.4.1.3. Cladding corrosion

3.3.1.4.1.4. Cladding stress

3.3.1.4.1.5. Cladding fatigue

3.3.1.4.2. Fuel Assembly Analysis

3.3.1.4.2.1. Fuel assembly growth

3.3.1.4.2.2. Fuel assembly structural corrosion

3.3.1.4.2.3. Fuel assembly normal operation stresses

3.3.1.4.2.4. Fuel assembly normal operation fatigue

3.3.1.4.2.5. Fuel assembly LOCA/Seismic stresses

3.3.1.4.2.6. Fuel assembly shipping and handling

3.3.1.5.LOCA Safety Analysis

This section is the testing of the LOCA analyses to ensure that in the cases of large or small break LOCA that adequate core cooling is maintained. No specific subsection have been identified other than SBLOCA and LBLOCA.

3.3.1.6.Non-LOCA Safety Analysis

This section is the testing of the non-LOCA analysis to ensure that it follows the requirements needed. No specific subsections have been identified.

3.3.2. Experience

For this section, all of the experience that has been obtained for testing, manufacturing, assembly, etc. will be compiled. It is important that the NRC see that the new fuel, cladding, or geometry will be evaluated and analyzed by those who are already familiar with the fuel fabrication process.

3.3.3. NRC Interaction

3.3.3.1. Evaluation of Revised Topicals

All of the topicals that are currently used to support zirconium clad, cylindrical UO_2 fuel will need to be changed so that they reflect the needs of the new fuel. These will include those changed for fuel performance, core physics, LOCA, and thermal-hydraulic analyses.

3.3.3.2. Acceptance of LTATR

This is the allowance of the LTATR for use in a commercial or research reactor by the NRC

3.3.3.3. Acceptance of LAR by licensee to allow LTATR

This is the acceptance of a license amendment request by the licensee for allowance of an LTATR inside of their commercial or research reactor

3.3.4. Technical Issues

The technical issues that are listed below have been based upon years of irradiation experience of many fuel assemblies. These issues have been essentially eliminated by manufacturing processes and a better understanding of the environment within the core. If changes to the structure of the fuel assembly do not occur (that is, only fuel and cladding changes) then debris and grid fretting are not new license issues. However, creep collapse, end plug weld, hydriding and pellet cladding interaction will be very important when analyzing all of the new licensing scenarios. Since these are already known failure mechanisms, they will need to be addressed before the NRC will allow for a full core load.

The next two subsections describe common fuel reliability failure mechanisms and current industry issues that are/were studied. The fuel reliability failure mechanisms are known to the NRC and will need to be addressed during the modeling and analyses for the new fuel, cladding and geometry.

3.3.4.1.Fuel Reliability – Failure Mechanisms

3.3.4.1.1. Debris Fretting – wear from particles in coolant

3.3.4.1.2. Grid Fretting – wear from fuel rods rubbing the grid

3.3.4.1.3. Creep Collapse – cladding losing its cylindrical shape

3.3.4.1.4. End Plug Weld – improper welding of end plugs

3.3.4.1.5. Hydriding – formation of metal hydrides at stress points

3.3.4.1.6. Pellet Cladding Interaction

3.3.4.2.Current Industry Issues

3.3.4.2.1. Incomplete Rod Insertion – inability to fully insert control rods

3.3.4.2.2. Axial Offset Anomaly – shift in the flux due to deposition on fuel rods

3.3.4.2.3. CRUD deposition – deposition of particulate matter on fuel rods

3.3.4.3.Other Technical Issues

This portion is to allow the addition of technical issues that have not been addressed above but are found to be important at a later date. This will be more important when results from the LTATR are analyzed.

3.4. Initial Confirmation through Lead Test Assembly with Test Rods

Once the analyses and models have been approved and validated and the NRC has granted permission for insertion of lead test assemblies with test rods, a course of action to manufacture, transport, store, load and analyze these rods must be completed.

3.4.1. Manufacturing

This will describe the process needed to manufacture the lead test assemblies with test rods that will be inserted into a few fuel assemblies for testing. Any concerns about the ability to produce high quality lead test assemblies with test rods will need to be addressed.

3.4.2. Transportation

This section will describe any potential problems associated with the transportation of fuel assemblies containing lead test assemblies with test rods. For instance, for a higher density fuel, does the increase in density of the fuel cause enough of an increase in reactivity to cause concern? Also, does the increase in density of the fuel or cladding cause enough of an increase in the weight of shipping loads to require a new container that would have to be licensed by the NRC?

3.4.3. Storage

This section will describe any specific needs for storage of the fuel assemblies containing lead test assemblies with test rods. These must address the ability to reliably store the spent test assemblies in a fuel pool, dry casks, and a long-term repository.

3.4.3.1.Fresh fuel

Unless the fuel is non-uranium based, there is no need to change the procedures already in place to ship and store fresh fuel. If the fuel is mixed with plutonium or some other radioactive material, then procedures to mitigate any additional radioactivity from the new fuel will need to be in place before shipments can occur. There should be no need to change this section for cladding or geometry.

3.4.3.2. Irradiated fuel during storage in a spent fuel pool, dry cask, or repository

Analyses will have to be performed to ensure that the lead test assemblies with test rods inside the fuel assemblies will not cause any problems to the loading of the spent fuel within the pool. The main analysis that must be performed is a fuel assembly drop accident. The purpose is to ensure that if the test assembly where dropped, it would no greater radioactivity release than could be expected if a normal fuel assembly where dropped. The acceptance of a non-standard fuel assembly by the government into the repository will have to be proposed to and evaluated by the NRC/DOE.

3.4.4. In-Core Loading Pattern

There will need to be analyses run to determine the proper placement of the fuel within the core. This is important to ensure that the power levels where the LTATR is placed within the core will not cause the LTATR to become the limiting fuel assembly for normal, transient and accident conditions. This may have previously been accomplished when the core physics analyses were performed.

3.4.5. Analysis of the LTATR data against developed models/analyses

Analyses of the LTATRs need to be performed to confirm the assumptions made in the models/programs and testing performed in the previous sections. This is accomplished by adding data to the models that were previously based only upon mathematical equations. The data will be gathered through post-irradiation examinations (PIEs) and evaluating the results of the PIEs against the results predicted by the models/analyses.

3.4.5.1. Post-Irradiation Examination

Post-irradiation examinations (PIEs) need to be performed on the lead test assemblies with test rods. There is possibility that some may need to be performed poolside and others will be shipped off-site for hot-cell examinations by an outside source with the capability to do so. These examinations will follow the guidelines that are available in the "Fuel Examination" section.

3.4.5.2. Evaluation of Examination Results

There will need to be evaluations of the data gathered from the PIEs. This data will be used to confirm the models/programs and other analyses that were performed prior to the acceptance by the NRC for insertion of the LTATRs. When this data confirms the models/programs and analyses performed earlier, the effort for NRC acceptance of the placement of Lead Test Assemblies (LTAs) will commence. The "Fuel Examination" section outlines the post-irradiation examinations that should be confirmed.

4. LTA Loading – Phase 2

Design, testing, and analysis of the new fuel rods and assemblies have already been accomplished at this point. Preliminary confirmation by testing of the models has also been established. The goal of this phase is the acceptance by the NRC for placement of Lead Test Assemblies into a commercial power reactor. Most of the previous analyses completed for the Lead Test Assembly with Test Rods will still be valid here. There will need to be an evaluation of the entire assembly with the new fuel, cladding, or geometry instead of only one or a few rods. Since there may not be much data gathered on the new fuel inside of light water reactor, it will be important to test these fuel assemblies in more than one reactor for at least a full fuel life cycle which my require up to 3 cycles of exposure.

It is expected that upon completion of this phase, the NRC will agree to full core loading of the new fuel. However, the NRC may also require further testing. If this occurs, the timeline for acceptance to the full core load will be extended. This has not been accounted for in the analyses that were performed for the timeline and cost of new fuels, cladding, or geometries.

4.1. Reasons for Lead Test Assembly

There are many reasons for the use of lead test assemblies when analyzing new nuclear fuels, cladding, and geometry. These reasons can be divided into three categories, confirmation of the models and analyses, confirmation of the manufacturing processes, and testing of the infrastructure needs. Upon completion of these, it is expected that full core licensing will be accepted by the NRC.

The confirmation of the models and analyses are most important when analyzing the irradiation history, burnup levels and neutronic response of the new fuel rods and assemblies. The outputs from the analyses, which are then confirmed by the post irradiation examinations, are the basis for acceptance by the NRC.

Confirmation of the manufacturing processes ensures that the vendor will be able to manufacture and assemble the new fuel, cladding or geometry within acceptable standards. This section is not important to the NRC but is very important to the vendor when they are trying to determine the actual costs and large scale procedures that will be needed to produce the new fuel. By taking the time to confirm the manufacturing process, any problems that could arise when taking this to full core loading can be addressed at an early stage. Confirmation testing will include minimizing the inclusion of impurities, and ensuring that the new fuel, cladding, and geometry will be fabricated within acceptable standards set by the fuel fabricator.

The testing of the infrastructure is the least important of the three categories. This is because unless the new fuel is non-uranium based, i.e., a mixed oxide fuel that contains plutonium, there is no need for any new procedures when transporting, inspecting, or storing. There may be some problems that could arise if the weight of the fuel assemblies were significantly increased due to an increase in the density of the fuel or cladding. The worst possible scenario would be that the fuel assemblies were too heavy for handling by the current fuel handling cranes. If this were to occur, it would require the licensee to reanalyze and modify the fuel handling cranes. This is currently not included in the timeline and cost structure. This is because it is a licensee task and not a fuel fabrication task. The weight of the fuel assembly should be determined early; therefore, if there needs to be changes to the licensee's fuel handling cranes, it will not add to critical path time and will be completed well in advance of irradiation.

4.2. Design Description of LTA

This will not change from of the design already completed for the testing of the Lead Test Assembly with Test Rods (LTATR). The main change will be that there will be a full assembly loaded with the new fuel instead of only a few rods within the entire assembly. The appropriate drawings/models will have to be revised. At this time the revisions are not seen as a major undertaking that will require much man power or time.

4.3. Fabrication of LTA

4.3.1. Pellet, Cladding, or New Geometry Manufacturing Process

This section contains the process that will be used to manufacture the new fuel assembly. This process should have a description, quality assurance program, feed material requirements, and pellet, cladding, and geometry qualification for production.

It is not believed by vendor engineers that there will need to be changes to the operating license of the fuel manufacturing plant if the U-235 content is kept below 5%. The operating license of the manufacturing plant will not need to be changed since the main requirement for their licensing is that U-235 content of their UF_6 precursor be kept below 5%. Therefore, lower density fuels may cause problems with this section, because of the possible higher enrichment needs. Licensing a new fuel fabrication facility is outside of the scope of this thesis and will not be included in the cost structure or timeline.

4.3.2. Fuel Rod and Assembly Manufacturing Process

A fuel rod and assembly qualification and production description are needed. This should include the process for analysis of fuel rods for failure mechanisms, i.e. missing pellets, poor end welds, etc.

4.4. Infrastructure

This will allow the fuel manufacturer to prove that it can transport, inspect, store and load the new fuel assemblies. This will be important for not only the introduction of the LTAs, but will also provide a "dry run" for full core implementation. It is acceptable that the infrastructure needs may change during full core loading. This will be due to the higher volume of fuel assemblies that will need to be produced.

4.5. NRC Approval

A new submittal of the topicals for fuel performance, core design, LOCA and non-LOCA analysis, and Thermal-Hydraulic analysis based on previous testing will be required. Since there is nothing changing except the specific scenario (fuel, cladding, or geometry), the experimental data found from the LTATRs should bound the expected behavior of the entire fuel assembly.

4.6. Irradiation Plan

A plan for the irradiation of the Lead Test Assemblies needs to be in place before they are inserted into the reactor for testing. Consideration should be taken to have a few assemblies irradiated to different burnup values. It should be possible to have one assembly removed at the first refueling outage, a second assembly removed for the second refueling and finally, two or more can be removed from the reactor after the third refueling outage. This will give data for the fuel at various burnups and more than one rod from each refueling outage can be analyzed. As noted above, significant change will require LTAs in more than one reactor.

The lead assembly location within the reactor will need to be determined with the following parameters. The rods will be located so that the test assemblies have a high enough power load that they will be representative of operating reactor conditions. Core design will determine the best/safest position to place the LTA within the core.

4.7. Fuel Examination

Post-irradiation examinations (PIEs) will allow for data to be gathered from the fuel assemblies for complete verification of the models and testing performed on the LTATRs.

4.7.1. Characterization of the Fuel Rod / Assembly Prior to Irradiation

This allows for a baseline when analyzing irradiated fuel rods. The initial measurements that need to be taken follow below.

4.7.1.1. Pellet

4.7.1.1.1. Grain Size
4.7.1.1.2. Microstructure
4.7.1.1.3. Resinter Test Performance
4.7.1.1.4. Diameter
4.7.1.1.5. Length
4.7.1.1.6. Porosity Distribution
4.7.1.1.7. Complete Chemical Impurity Content

4.7.1.2. Fuel Rod

4.7.1.2.1. Length of Rod
4.7.1.2.2. Pellet Active Length
4.7.1.2.3. Plenum Length
4.7.1.2.4. End Plug Welds
4.7.1.2.5. Weight of Pellets

4.7.2. Poolside PIE

After each cycle, the assemblies will be examined at the poolside using non-destructive examinations. These examinations include

4.7.2.1. Fuel Assembly Visual Examination
4.7.2.2. Fuel Rod Visual Examination
4.7.2.3. Fuel Rod CRUD Measurements
4.7.2.4. Fuel Rod Growth
4.7.2.5. Fuel Assembly Growth
4.7.2.6. Fuel Assembly RCCA Drag Force
4.7.2.7. Fuel Rod Oxide Thickness
4.7.2.8. Fuel Rod Fission Gas Release

4.7.2.9. Fuel Rod Bowing

4.7.2.10. Grid Width

4.7.2.11. Grid Oxide Thickness

4.7.2.12. Guide Thimble Plug Gauge

4.7.2.13. Guide Thimble Oxide

4.7.2.14. Fuel Assembly Bow and Distortion

4.7.2.15. Clad Creepdown

4.7.3. Rod Extraction and Hot Cell Examinations

After each of the cycles, some of the rods will be removed, and hot cell examinations will be performed.

The minimum amount of analyses to be performed is as follows:

4.7.3.1. Fission Gas Release

4.7.3.2. Fuel Clad Metallography

4.7.3.3. Fuel Pellet Ceramography

4.7.3.4. Pellet-Cladding Interaction

4.7.3.5. Burnup Analysis

4.7.3.6. Burnup Distribution

4.7.4. Operational History

Detailed data will be gathered from all of the analyses into a database to allow for evaluation of the lead test assemblies performance versus model prediction. It is suggested that one of the assemblies be placed in an instrument location to verify predicted neutronic performance. During irradiation, power levels, temperature, transient conditions, and RCS chemistry needs to be recorded in detail.

4.7.5. Acceptance Criteria

Acceptance criteria will need to be developed before the irradiation of the fuel assemblies is complete. The criteria will come from the fuel vendor with final acceptance by the NRC since the NRC will determine what will be required to be analyzed. Analysis of the fuel assembly after the LTATR irradiation may also indicate that further testing needs to be performed on the LTAs during their PIEs.

Upon verification that the fuel assemblies were within the limits set by the acceptance criteria for post-irradiation fuel assemblies, and the models and analyses were updated and accepted by the NRC, the program will continue to the full core loading phase.

5. Full Core Loading – Phase 3

This phase allows for the determination of the best approach to the demands that will be required if widespread use of the new fuel, cladding, or geometry is accepted by the utilities.

5.1. Production of Assemblies

This is again the chance to verify that the production process used for fuel, fuel rods and assemblies are optimized. From this analysis, the best procedure to manufacture the fuel, fuel rods, and assemblies will be determined.

5.2. Shipping, Handling, and Storage

Previously any special needs for the shipping, handling, and storage of a few lead test assemblies were evaluated. Any special procedures that were developed will probably remain applicable for the full scale production, however, they will need to be reanalyzed to prove that they will work for the large volume of fuel assemblies that will be shipped, transported, and stored.

5.3. Spent Fuel Pool

Analyses performed on the fuel for spent fuel pool storage and for the load dropping accident that would damage the new fuel assemblies will have already been completed for the LTA assessment. During the LTA analysis, the ability to store the fuel in the spent fuel pool should have been determined and any other problems associated with performance of the fuel assemblies in the pool should have been addressed.

5.4. Permanent Disposal

Long-term storage requirements outside of the spent fuel pool in a national spent fuel repository are not yet established. Currently only uranium dioxide clad in a zirconium alloy and governmental waste has been analyzed for disposal. There will need to be a change to the current analyses to allow for the inclusion of the new fuel and cladding into the repository upon its completion.

5.5. Security and Safeguards

Since enrichment of the uranium inside the new fuel will be no more than standard UO_2 fuel, there is no need for additional security to be established. The fuel will be well below the enrichment needed for nuclear proliferation, and is considered low enriched uranium and should not involve any further complications for the licensee. The changes to the cladding or geometry will have no effect on the security or safeguards that need to take place.

5.6. Irradiation of Full Core and PIEs on Fuel Assemblies

At this point, the irradiation of the full core will take place. After the fuel has been irradiated, several representative fuel rods will be extracted from an irradiated fuel assembly at each refueling outage. These fuel rods will be subject to poolside and hotcell examinations (PIEs) to provide further data and to confirm the models. It is expected that poolside and hotcell examinations will occur for all three refueling cycles that the fuel is expected to endure.

5.7. Evaluations of the Models and Analyses with Data from the Full Core Load

The data gathered from the PIEs performed during the various refueling cycles will allow for further refinement of the models and analyses. This data will confirm the upper bound and best estimate curves of all of the pertinent analyses and models evaluated during the core performance section. After these models have been updated, they will be sent to the NRC for acceptance. At this point, the application for rulemaking changes will be submitted.

5.8. Rulemaking

Rulemaking is the final step in full core implementation. This step will allow the new fuel, cladding, and/or geometry to be acceptable for all commercial power reactors and not just those that were in the initial study for LTA and full core implementation. At this point, changes to the FSAR of the licensee should be minimal which will decrease the costs associated with changing from the standard fuel assembly to the new fuel, cladding and/or geometry. This is very important since there are over 100 commercial nuclear reactors in the United States. If all were to request license amendments at the same time for the new fuel, it would lead to a backlog of work for the NRC which would hurt the fuel fabrication company's ability to sell the new fuel assemblies.

6. Phase 4 – Increased Performance above Standard Fuel Assemblies

It is expected that upon completion and acceptance of the full core load of new fuel, cladding, and/or geometry that an increase in performance above that achieved with standard fuel assemblies is the next logical progression.

One such increase in performance would be the ability to increase the established NRC burnup limits (62 GWD/t). For instance, it is conceivable that greater than 100 GWD/t burnup could be achievable with the proper fuel and cladding combination. The logical choice is to move to a higher density fuel and a more irradiation resistant cladding. On the other hand, a change to the geometry of the fuel would not likely increase the allowable burnup, but could increase the power per volume performance. Therefore, the changes to geometry that will be evaluated only consist of how the new geometry will react to a much higher burnup. For example, if fins are added, will the fins become dislodged during high burnups and now pose a loose particle risk. The sections below discuss the example of extended burnup limits.

6.1. Reevaluation of the Models and Analyses from 62 GWD/MTU to 100 GWD/MTU

The models that were previously built for evaluations up to 62 GWD/MTU will need to increase their limits on burnup. If the new fuel, cladding, or geometry is going to require large changes to the existing modeling and analyses, it may be valuable to add this portion in after the LTA or full core PIEs have been complete. This will allow for minimal backtracking upon work that is already performed. It is believed that as few as one to two more engineers per model or analysis is enough to make these changes. The time and cost for these additions can be seen in the "Timeline and Cost Structure" portion of this thesis.

6.2. Analysis of Fuel / Cladding Combination

If at this point the fuel and cladding have been accepted for full core loading separately, it will pose less of a problem when the combining the two. If the new fuel and cladding are being developed simultaneously, then it should not require much extra effort to ensure that the models include both. For example, the fuel performance software should be able to analyze UO_2/zirconium alloy, UO_2/new cladding, new fuel/zirconium alloy, and new fuel/new cladding. By doing this at the beginning, it will increase the costs during the beginning stages, but should decrease the modeling and analysis needed during later stages.

6.3. Lead Test Assembly with Test Rods / LTA

Lead test rods and lead test assemblies will again have to be fabricated upon completion of the analyses. With the data already gathered from the initial irradiation of the new cladding and new fuel, it may be acceptable to the NRC to skip the LTATR and go directly to the LTA. This is not unprecedented, for MOX fuel Framatome/ANP skipped test rods and went directly to LTAs because of the extensive European use of MOX fuel and therefore subsequent histories and performance data.[6]

The LTATR and LTAs used for the new cladding may be able to be irradiated longer. This is because the limiting burnup for a fuel assembly depends usually on the cladding. This would only be available if the rod internal pressure were not too great. If the rod internal pressure were too great, then further irradiation of the new cladding would not be possible.

6.4. Full Core Loading with the New Fuel and Cladding for 100 GWD/MTU

This scenario will occur if the models, analysis, and PIEs all support the case that the new fuel/cladding combination can operate at this high of a burnup. Further PIEs will need to be performed on LTATRs/LTAs/Full Core Loads for to determine the extent of fission gas release and cladding degradation. This will allow for rod internal pressure, radioactive release data, and the overall state of the cladding to be reported. Usually the limiting criteria for a fuel rod are rod internal pressure and cladding oxide formation. If

[6] Framatome/ANP-DCS "Fuel Qualification Report" 2001

these criteria can be limited or eliminated (as in the case of oxide formation); then the acceptance of this new fuel/cladding combination will take much less time.

6.5. Continuing Examinations

Even after the PIEs are performed on the full core load, further poolside non-destructive examinations will be conducted to further evaluate the integrity of the cladding. These will be performed in accordance to the guidelines laid out in the "Fuel Examination" section. These further examinations will only add to what should be a vast amount of data gather from the initial PIEs for the LTATR, LTA, and full core load.

6.6. Rulemaking

After all of the previous steps have been completed, an increase from the current 62 GWD/MTU will need to be accepted by the NRC. This will lead to the same benefits that were previously described for the "Full Core Loading," "Rulemaking" subsection.

Chapter 4 – Timeline and Cost structure

The timeline and cost structure were built upon the licensing strategy laid out in the previous sections. The fuel, cladding and geometry licensing approaches each have their own individual timeline and cost. A fourth timeline and cost structure for the development of a combined fuel/cladding licensing approach was evaluated independently of the other three approaches. The timelines were developed through three different sources. The first is real experience that was gathered from Westinghouse employees who have worked on modeling and licensing for various projects. The second source is from NRC documents. By looking first at the submittal dates and then following the documentation until it is complete, the general time needed for answering NRC questions and a probable timeline for licensing can be determined. NRC questions are posed as requests for additional information (RAIs) and can be followed by the back and forth question and answers that go on between the NRC and the licensee. The final source for the timeline was the length of time that Framatome/ANP used when determining specific milestones for MOX fuel. These milestones included the initial planning of their modeling and licensing needs.

The rates used were as follows: \$150/hr for a vendor engineer, \$220/hr for NRC engineering time, and \$5,000,000 for a PIE. The PIE includes the cost to remove the fuel assembly or assemblies, and transportation to a hotcell facility and analysis. Engineering time to analyze the results and update the models was calculated separately. Note, the costs were fixed and were not adjusted for inflation in the later years of the cost analysis.

When models were completed or updated and then sent out to the NRC, the amount manpower assigned to the engineering support was double that originally assigned to the NRC. This is deemed acceptable because when the NRC reviews items, it sends out RAIs that must be answered. Since NRC review time is critical path, it is only reasonable that answering RAIs should have increased manpower in order to keep the amount of critical path time to a minimum. The only variance on this is during the rulemaking task. Since most of the time needed for rulemaking is waiting for RAIs from the NRC, the amount of NRC time needed is assumed to be 6 months over the course of 2 years. However, since this will be a very active time for evaluation of the RAIs, the engineering support was set at 4 engineers for the 24 months allotted for rulemaking. The reason for the number of engineers is that it allows for 1 engineer to be assigned to each of the four core design analyses. This time will be used to not only answer RAIs but to also send personnel to the NRC headquarters in Maryland for the public rulemaking hearings.

Table 3 lists the expected dates for the fuel, cladding and geometry licensing approach milestones to be reached. These are the comparable modeling, irradiation and NRC approval milestones that were evaluated to determine an overall timeline. These milestones are almost all of the critical path components.

Table 3 - Milestones for Fuel, Cladding and Geometry Licensing Approaches

Milestone	Fuel	Cladding	Geometry
Phase 1			
Analysis of Standard Fuel Assembly	34th month	34th month	34th month
Evaluation of Fuel and Core Design	36th month	45th month	50th month
NRC Topical Report Review	47th month	56th month	66th month
Acceptance of LTA with Test Rods	50th month	59th month	69th month
Irradiation Inside Test Reactor Complete	56th month	65th month	75th month
Phase 2			
NRC Approval for LTAs	63rd month	70th month	87th month
Irradiation Cycle 1	80th month	87th month	104th month
Irradiation Cycle 2	98th month	105th month	119th month
Irradiation Cycle 3	116th month	123rd month	137th month
NRC Evaluation of Data and Revised Models	129th month	137th month	159th month
Phase 3			
NRC Approval for Full Core Loading	132nd month	138th month	162nd month
Irradiation Cycle 1	148th month	155th month	179th month
Irradiation Cycle 2	166th month	173rd month	187th month
Irradiation Cycle 3	184th month	191st month	205th month
NRC Evaluation of Data and Revised Models	197th month	204th month	246th month
Changes to Rulemaking Accepted	219th month	227th month	262nd month

From the milestones, the most important marker is the approval for full core loading. At this point, the NRC has accepted that the changes to the standard fuel assembly are safe and will not pose a danger to the public. The remaining PIEs are to evaluate the changes under full core conditions and to further refine the models so that they are consistent with the most accurate data available. The acceptance of rulemaking is to ensure that there is not a constant need by the licensee to apply for changes to their FSAR. This can be costly to the licensee and would reduce the economical advantages when $100,000 to $200,000 is spent every time a commercial reactor is refueled.

The cost structure for the three scenarios has been divided into a few sections. First, the overall cost of the scenario and the percentage that each resource uses was calculated. Next, the overall cost per year was calculated and finally, the number of man years for vendor and NRC Engineers were calculated.

The overall time and cost for each scenario is shown in Table 4. This combines the cost for vendor engineers to do models, analyses, and NRC engineering support; NRC engineers to evaluate, comment on, and issue RAIs, approval and rulemaking; and finally, the cost for each PIE. As previously stated, the costs for a PIE are fixed and each scenario used seven PIEs for complete analysis. Since the PIE costs are fixed, only the vendor and NRC engineering costs differed for each scenario.

Table 4 - New Fuel, Cladding, and Geometry Cost Structure and Time to Completion

	Fuel	**Cladding**	**Geometry**
Overall Cost	$60.75 Million	$67.46 Million	$90.48 Million
Time to Finish	18 Years, 3 Months	18 Years, 11 Months	21 Years, 10 Months
Time to Full Core Load	7 Years, 11 Months	10 Years, 6 Months	13 Years, 6 Months

The overall cost per year is evaluated in figures 4, 8, and 12. These three figures show the cost to perform work for each year in the licensing approach. These are probably the most important figures because they give how much the budget needs to be for each year. From this, planning can be done and a determination of how much, if any, outside money needs to be raised.

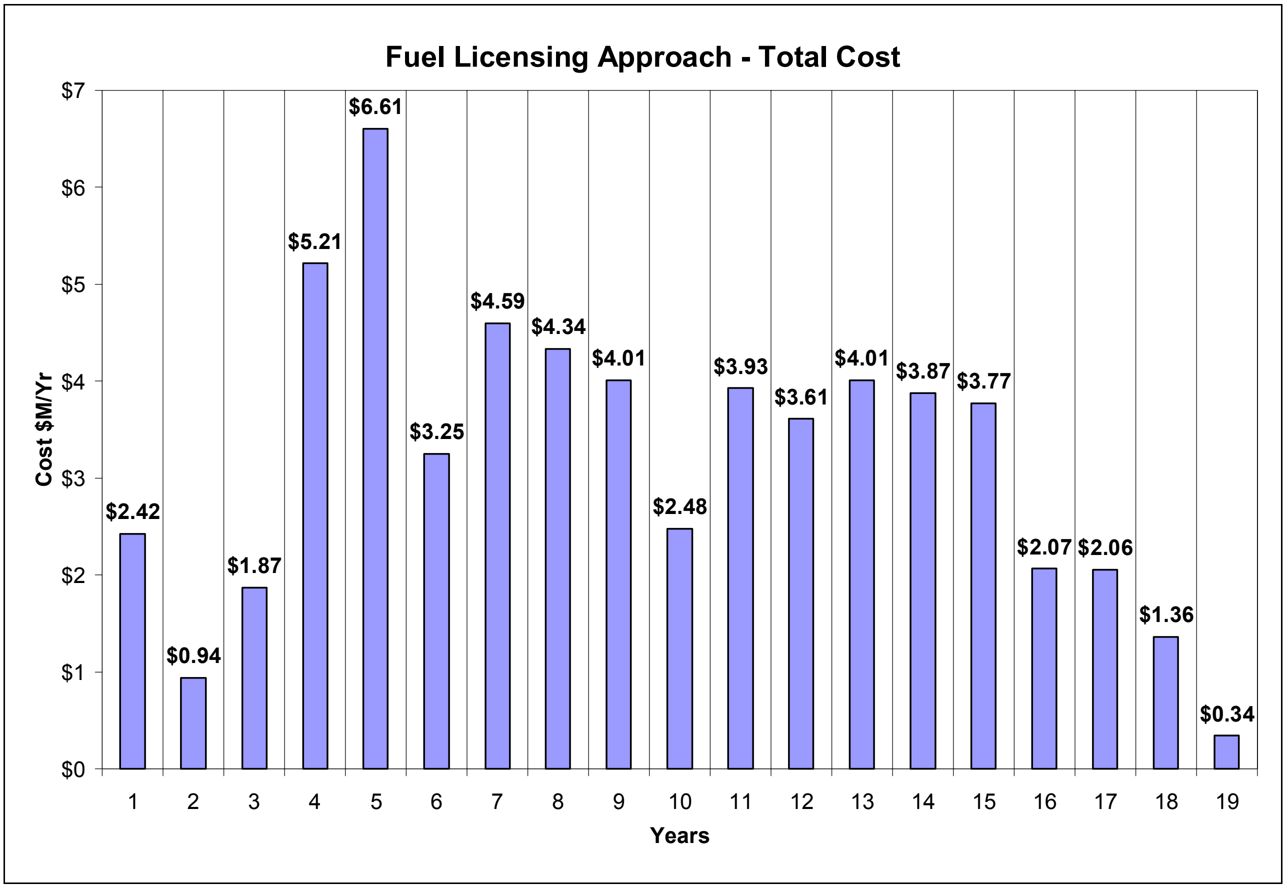

Figure 4 - Cost per Year - Fuel Licensing

The overall cost per year for the fuels licensing is found to never exceed $7 million. The costs for one year are below $4 million for almost 75% of the fiscal years. The maximum occurs in 2010 because the entire $5 million cost for the PIE of the LTA with Test Rods must be completed. Looking at Figure 5, the percentage that each resource uses is shown. Then in Figure 6, a break down of each resource per year is shown. By comparing the costs for resources, it is seen that the cost of engineering support is only 43% of the total or approximately $25.75 million spread over 18 years. The concentration of engineering support is found to be in the first 3-4 years and then significant resources are not needed again until after the irradiation of the LTAs and full core load during reanalysis of the models. There will initially need to be upwards of 16 engineers working during the first 4 years, after that the load decreases to an average of about 6-8 man years per year. There is a final need for 4 man years per year during the final 7 years. There is one year (year 12) where no engineering support is needed. This is because during the first cycle of the full core load, the 18 month time span stretches from late in year 11 to early in year 13 and during this time there is no real need for engineering support. The ability to take advantage of this time will be discussed in the conclusions and recommendations.

The actual manpower needed per year is shown in Figure 7. This figure again reiterates the fact that no engineering support is needed in year 12. The bulk of the costs

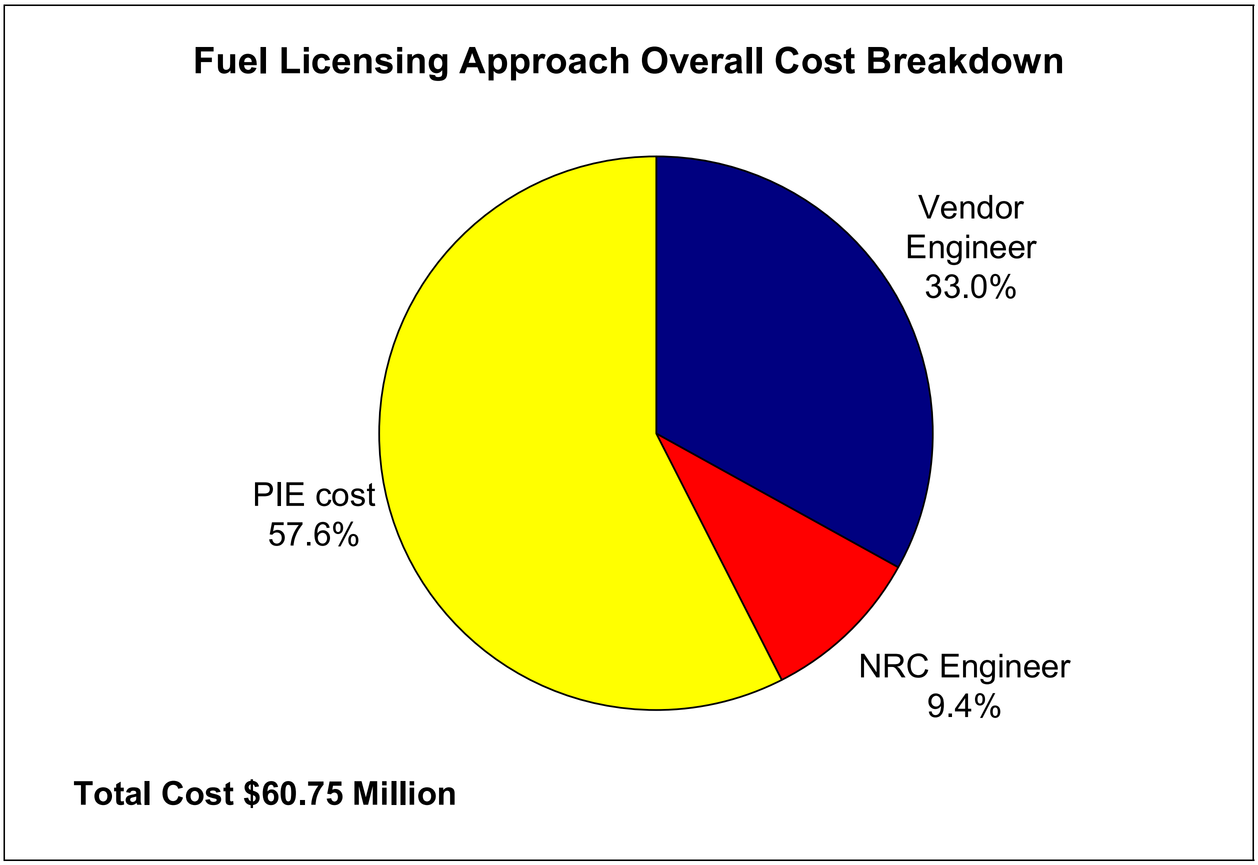

Figure 5 - Fuel Licensing Overall Cost Breakdown

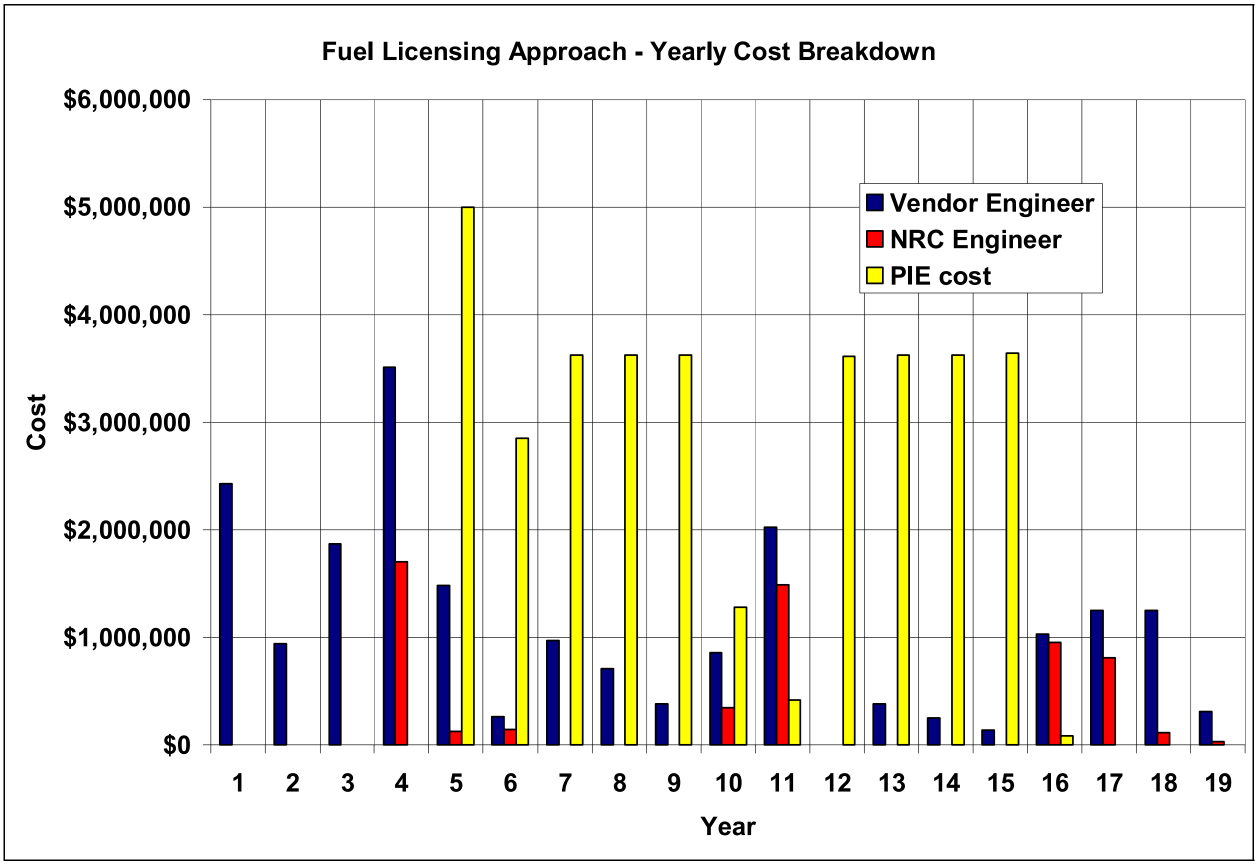

Figure 6 - Fuel Licensing Yearly Cost Breakdown

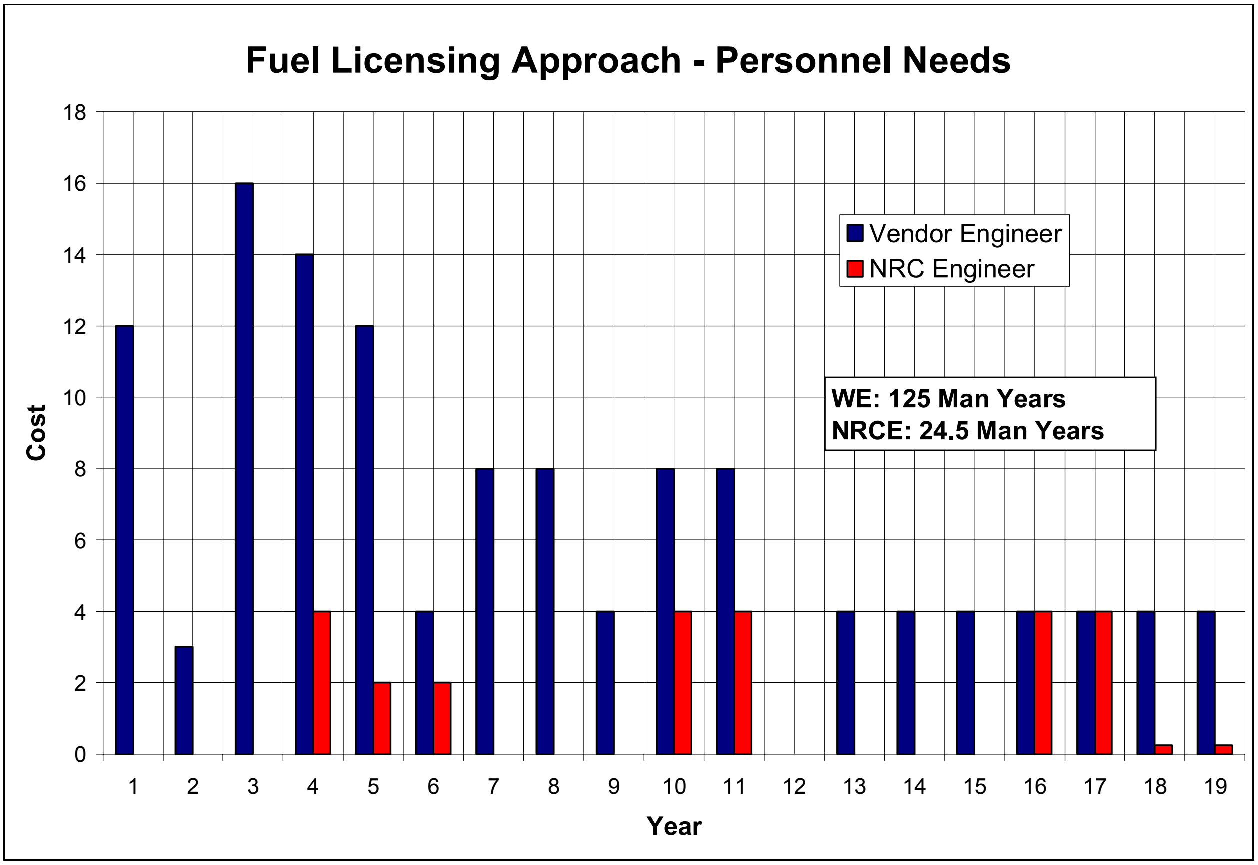

Figure 7 - Fuel Licensing Personnel Needs

Figure 8 shows the overall cost per year for the cladding approach. The total cost for any one year never exceeds \$7.2 million with the next highest cost being less than \$5.2 million. The remainder of the time spent costs approximately \$4 million or less. The year with the highest cost is 2010 and this is due to the entire charge of \$5 million for the LTA with test rods being due. Figure 9 illustrates the portion of the costs each of resources requires. Again, the total PIE cost is \$35 million, but, it has decreased to 51.9% of the total cost. The engineering costs are now 48.1% or about \$32 million. Similarly to the fuel approach, the Vendor engineering costs are prevalent in the first 4-5 years. Unlike the fuel approach, there are 2 years where no engineering support is needed for the cladding approach. These years, 7 and 13, preclude an increase to 4-8 man years of Vendor engineering support that will be needed. The Vendor support will be used during Phase 2 and Phase 3 for updating models and analyses with the data gathered from the PIEs.

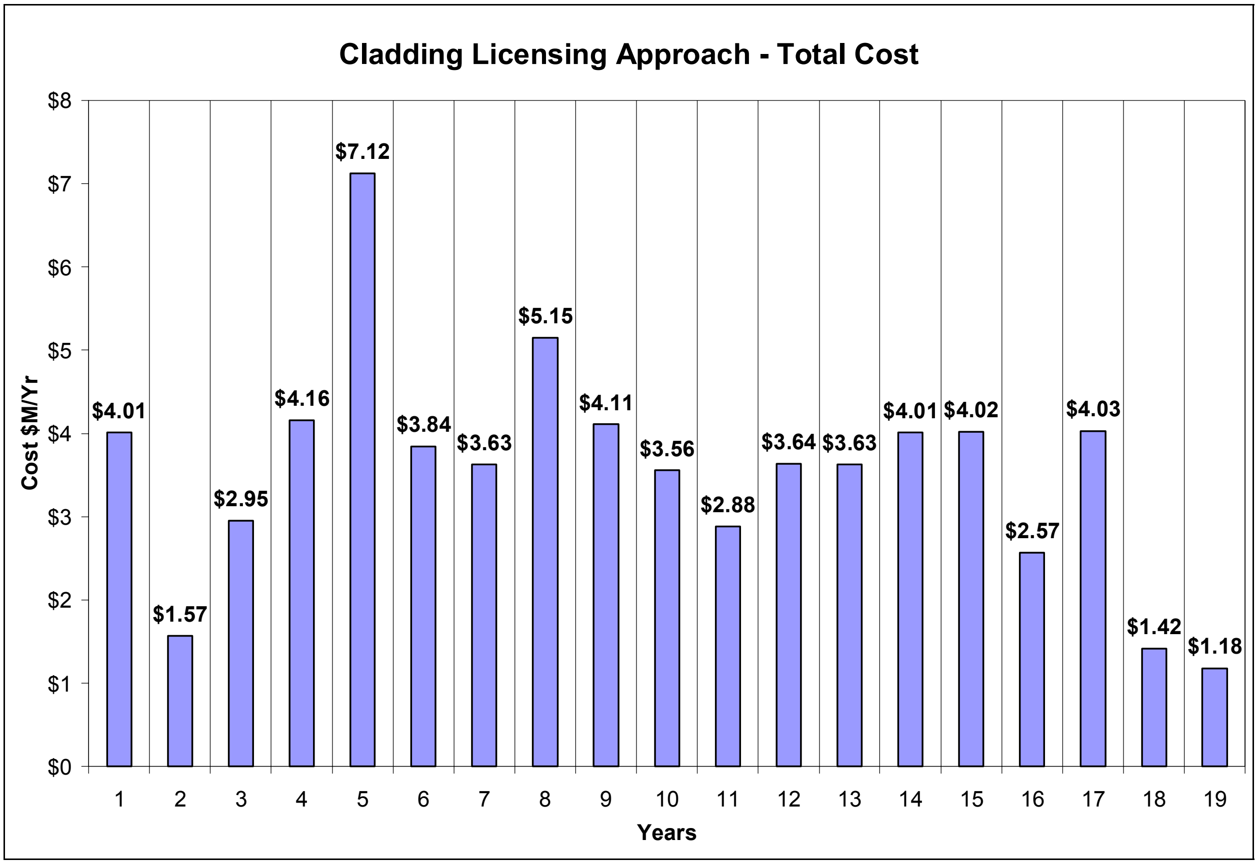

Figure 8 – Cost per Year – Cladding Licensing

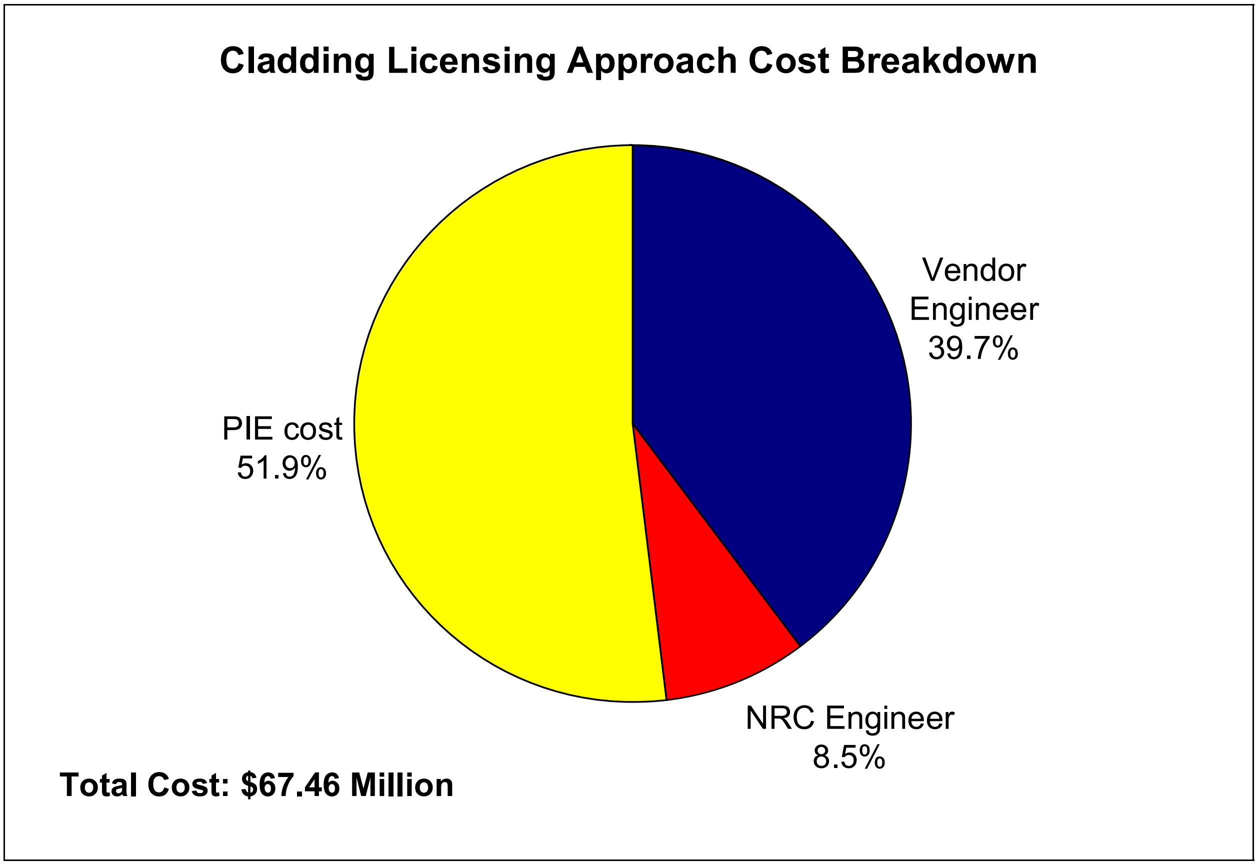

Figure 9 - Cladding Licensing Overall Cost Breakdown

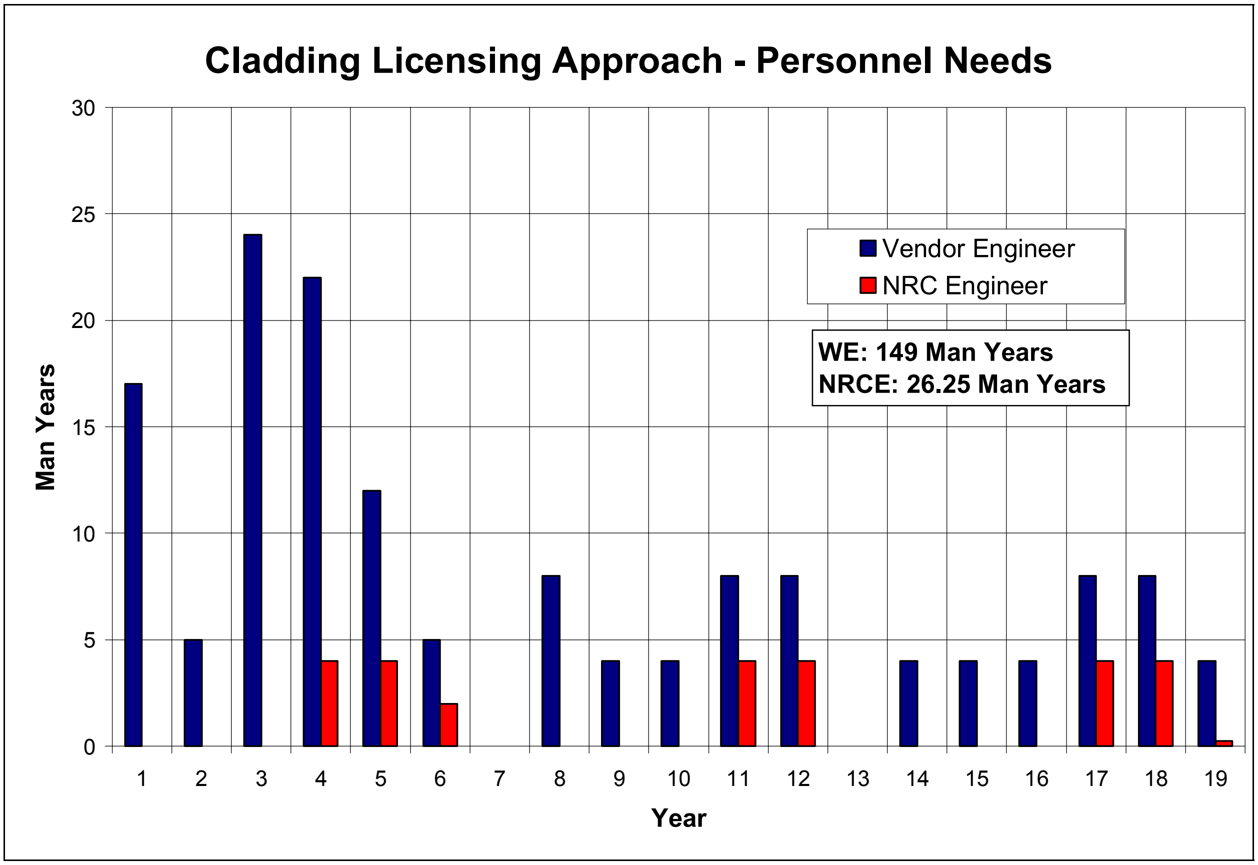

Figure 10 - Cladding Licensing Personnel Needs

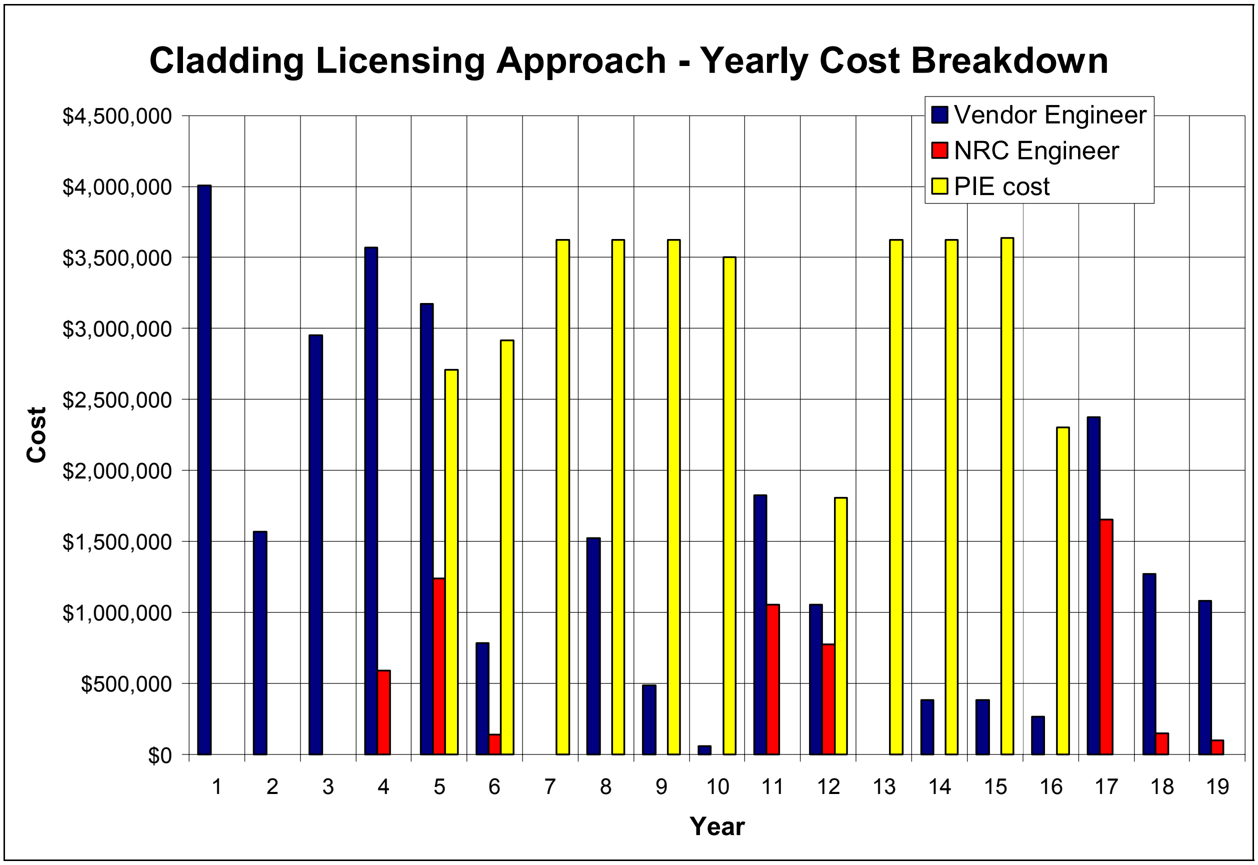

Figure 11 - Cladding Licensing Costs Yearly Breakdown

Figure 12 illustrates the overall costs per year for the geometry approach. The total cost for any one year never exceeds $10.5 million with the next highest cost being less than $7.5 million. The remainder of the time spent costs approximately $5 million or less. The year with the highest cost is year 4 and this is due to the large amount of engineering work for the modeling and analyses. Figure 13 illustrates the portion of the costs each of resources requires. Again, the PIE cost is $35 million, but, it has decreased to 39% of the total cost. The engineering costs are now 61% or about $55 million. Similarly to the fuel and cladding approaches, the Vendor engineering costs are prevalent in the first 4-5 years (Figure 14). There is one year (2020) where no engineering support is needed for the geometry approach. Figure 15 illustrates again the large need of engineering support in the first 4-5 years. This support hits a maximum of 40 persons for year 4. After the initial massive need for engineering, the personnel needs reach a point of approximately 4-8 person years for the remainder of the licensing. This support is to update the models and analyses after Phase 2 and Phase 3. At this point the work will consist of updating models and analyses from the data gathered during the PIEs.

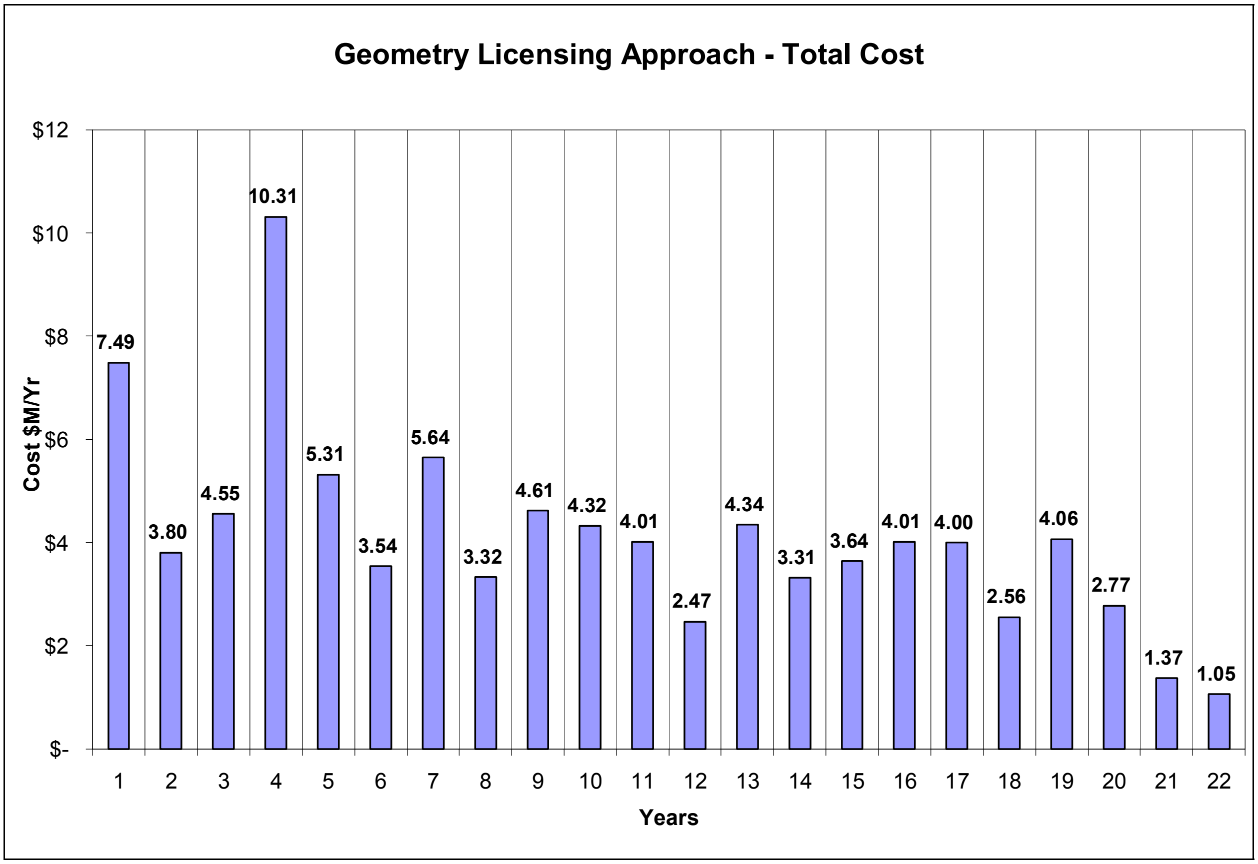

Figure 12 - Overall Cost per Year - Geometry Licensing

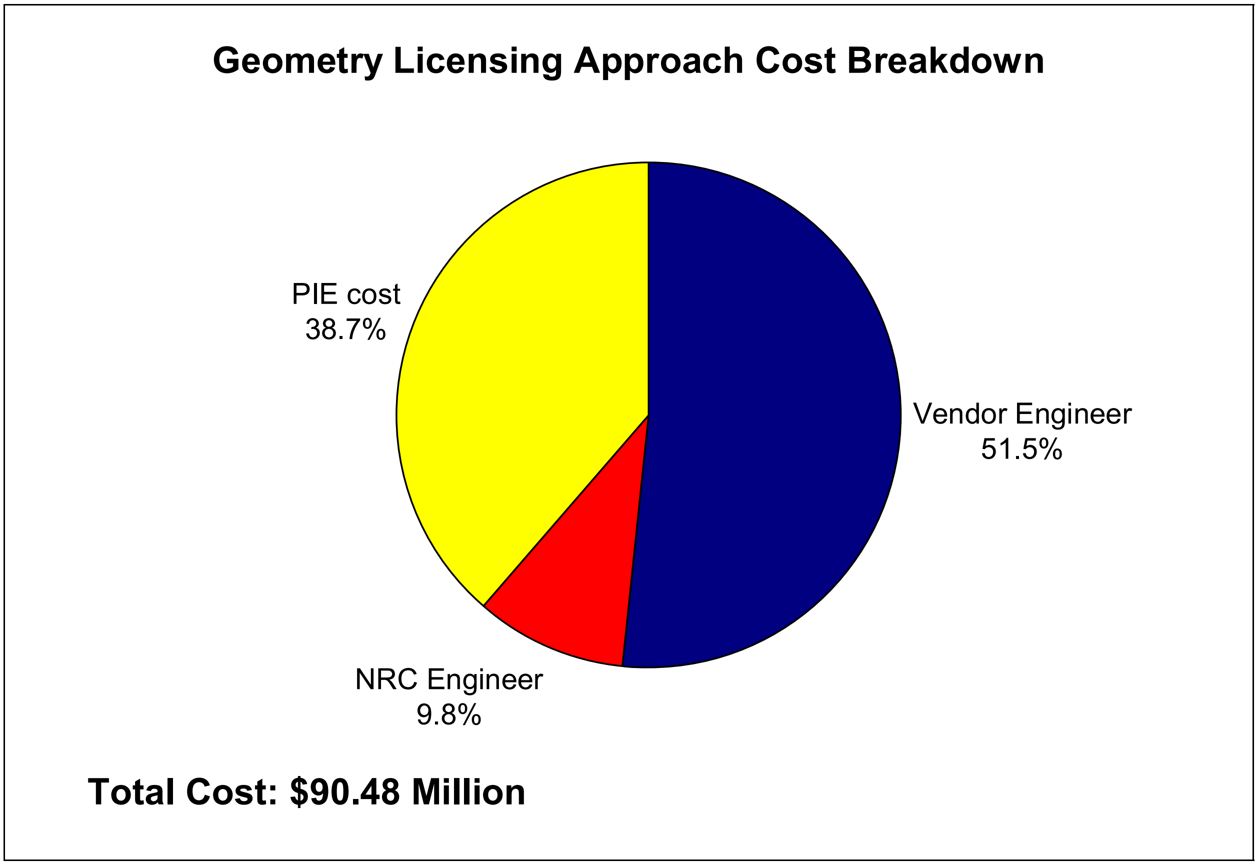

Figure 13 - Geometry Overall Cost Breakdown

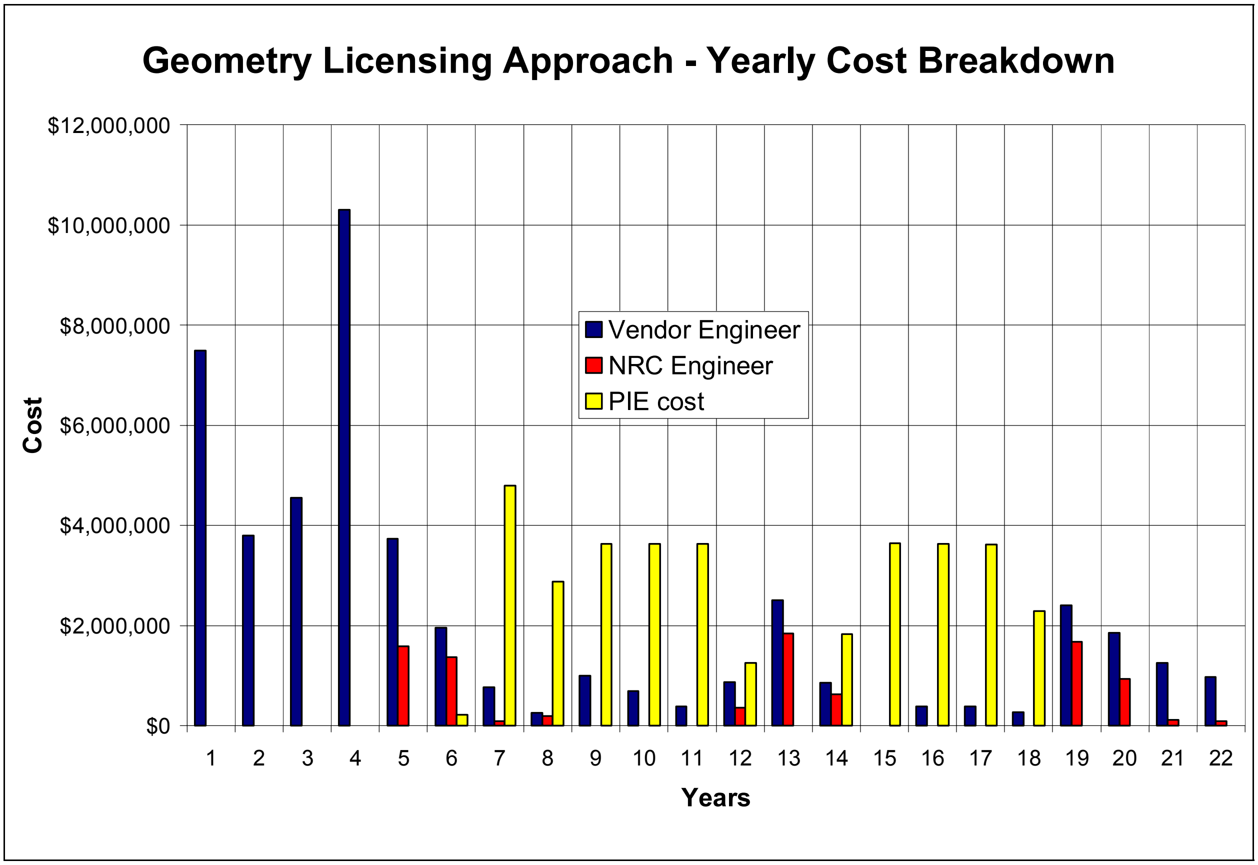

Figure 14 - Geometry Licensing Yearly Breakdown

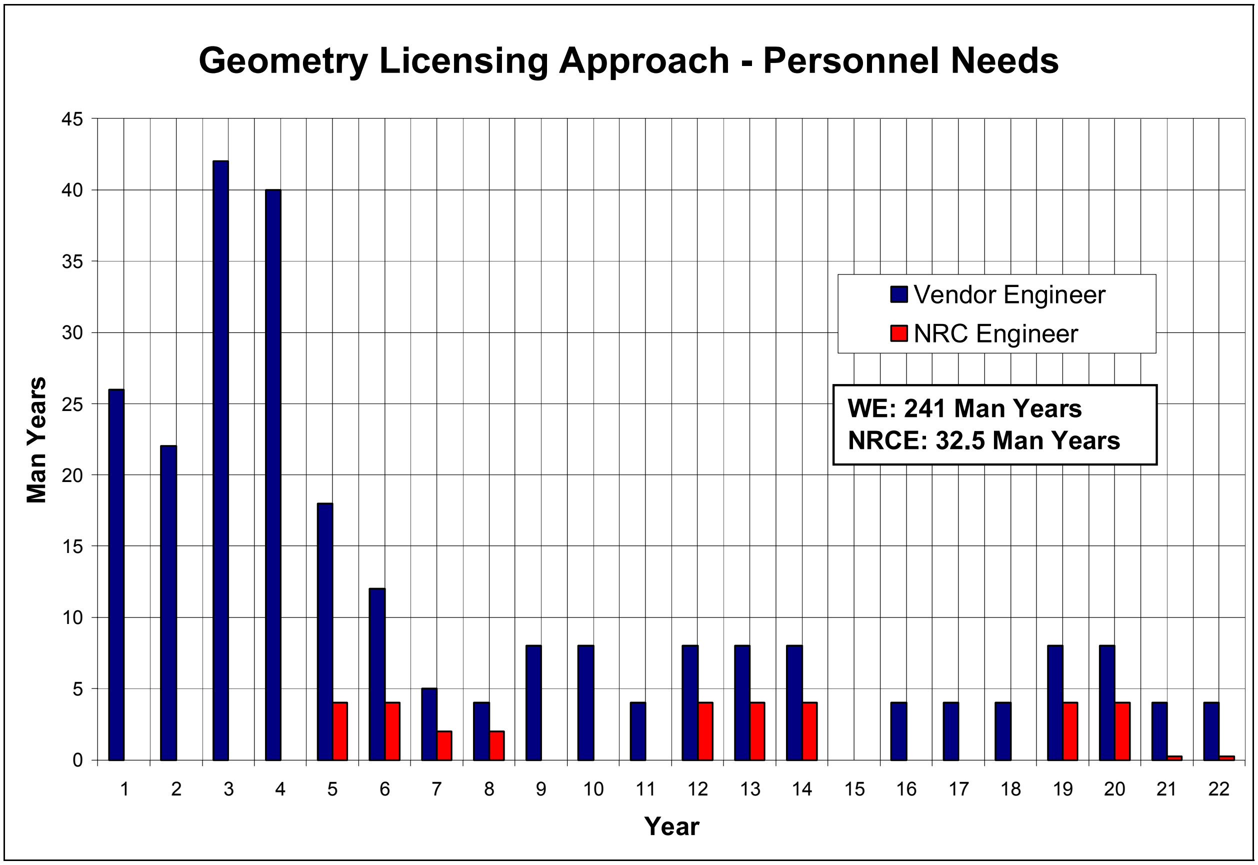

Figure 15 - Geometry Licensing Personnel Needs

The combined fuel/cladding approach saves almost $7.7 million ($120 million vs. $128 million) and does not increase the timeline more than a few months (3 extra months to reach full core loading) when compared to a simultaneous licensing of the fuel and cladding. This is possible because many tasks were combined and the critical path tasks had extra personnel assigned. The reason it is possible to combine the fuel and cladding tasks is because they are working on essentially the same modeling and analyses. The only difference is that one is looking at the fuel and one is looking at the cladding. If the synergistic effects of the fuel/cladding combination are to be realized within a reasonable amount of time, then by combining them together, the total amount of man power is reduced and the total overall cost is reduced. For example, comparing Figures 7 and 10 with Figure 16, we get a new figure, Figure 17. This figure displays a cost comparison of the combined vs. simultaneous approaches. From this figure, it is easily identifiable that except until the very end, it is cheaper every year to combine the fuel and cladding licensing. Savings are especially seen in the years 4 and 5 where the savings are almost $7.3 million for the combined approach. If the costs could be trimmed for the last two years of the combined approach, it is reasonable to believe that there could be a total savings of almost $10 million.

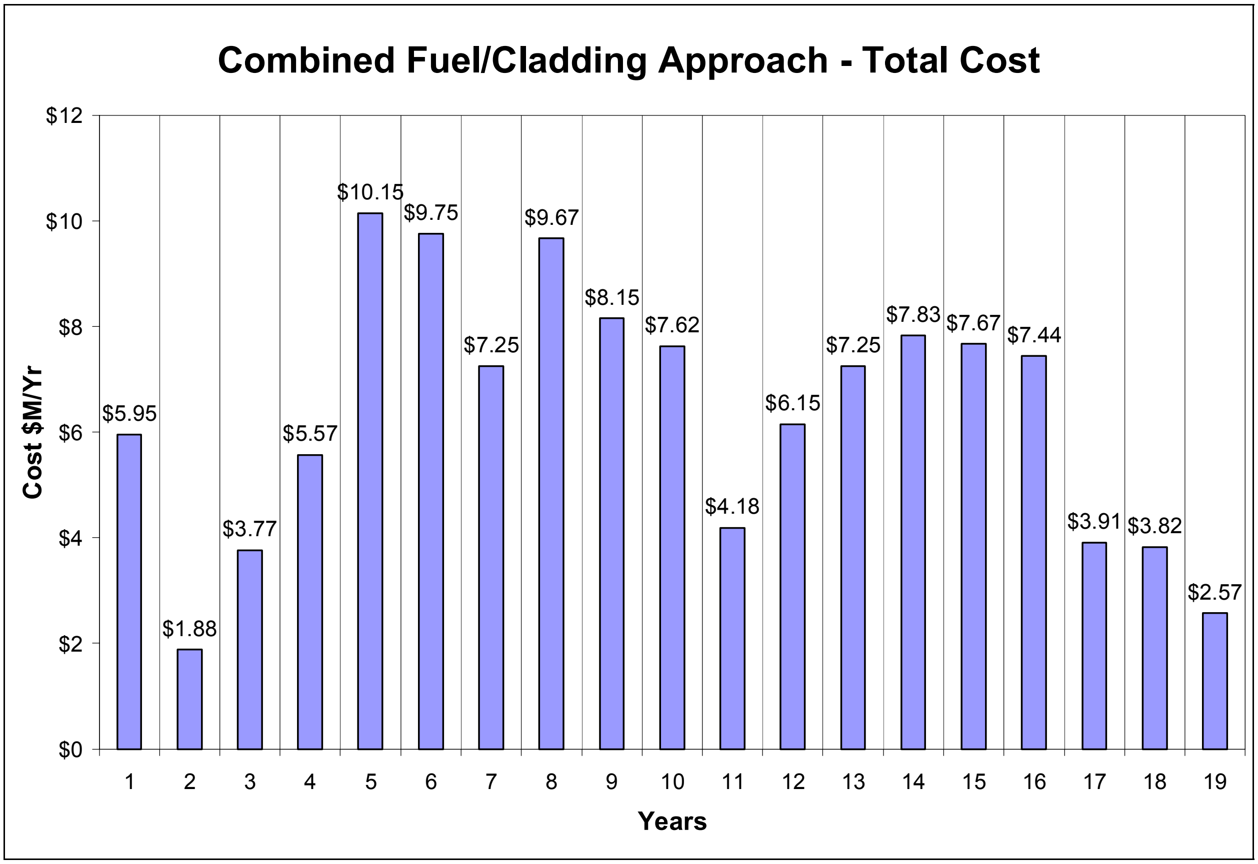

Figure 16 - Cost Per Year - Combined Fuel/Cladding Approach

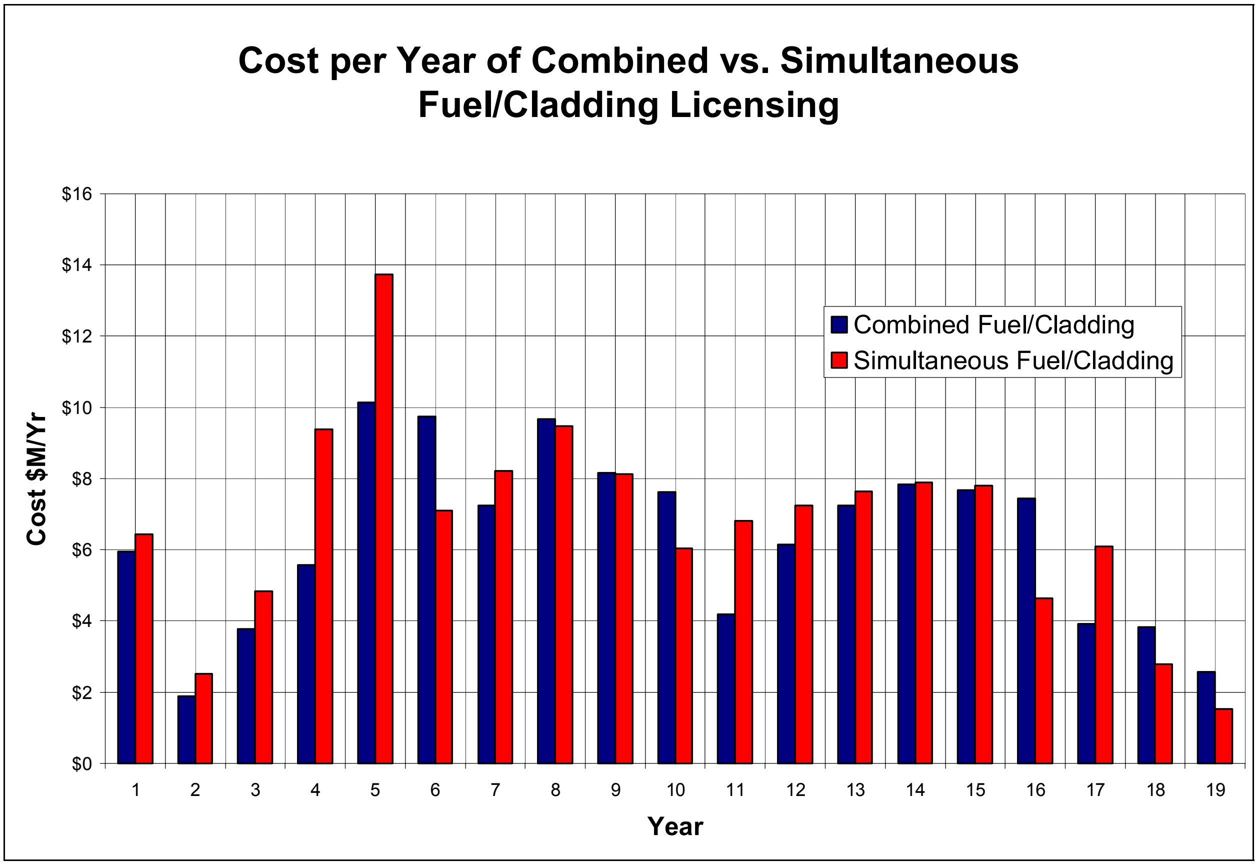

Figure 17 - Cost per Year Comparison of Combined vs. Simultaneous Fuel/Cladding Licensing

Another advantage to the combined licensing is that the maximum manpower needs for each year are smaller. This translates to a total of almost 60 man years less for the combined over the simultaneous licensing. Figure 18 illustrates how the manpower needs for the combined fuel/cladding approach are greatly diminished for the initial 4-5 years over that needed for a simultaneous approach. There are two years where no manpower is needed at all for the combined approach. For the simultaneous approach, there is always a need for manpower and never does the manpower needs ever go to zero.

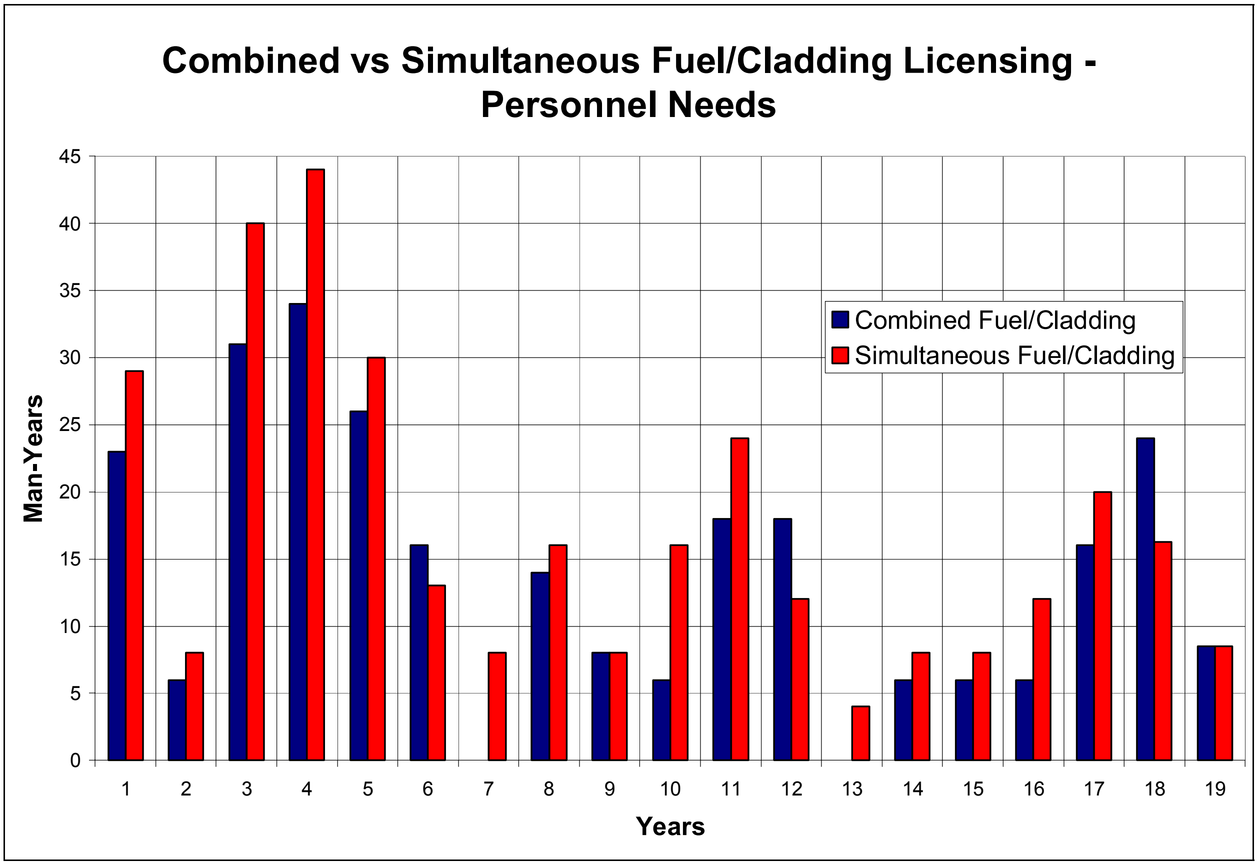

Figure 18 - Personnel Comparison for the Combined vs. Simultaneous Fuel/Cladding Licensing Approach

The final analysis performed was a combined fuel/cladding approach that first licensed both individually, then combined the two and increased the maximum burnup from 62 GWD/MTU to 100 GWD/MTU. Two scenarios were analyzed for the increased burnup. The first did not make an attempt to shorten the amount of time needed for licensing the increased burnup (Expanded scenario), and the second began Phase 4 when the second irradiation cycle of the full core for the new fuel and cladding was started. This was known as the Condensed scenario. From these, the overall costs were to remain the same, however, the total cost per year after the first 8 years varied (Figure 19).

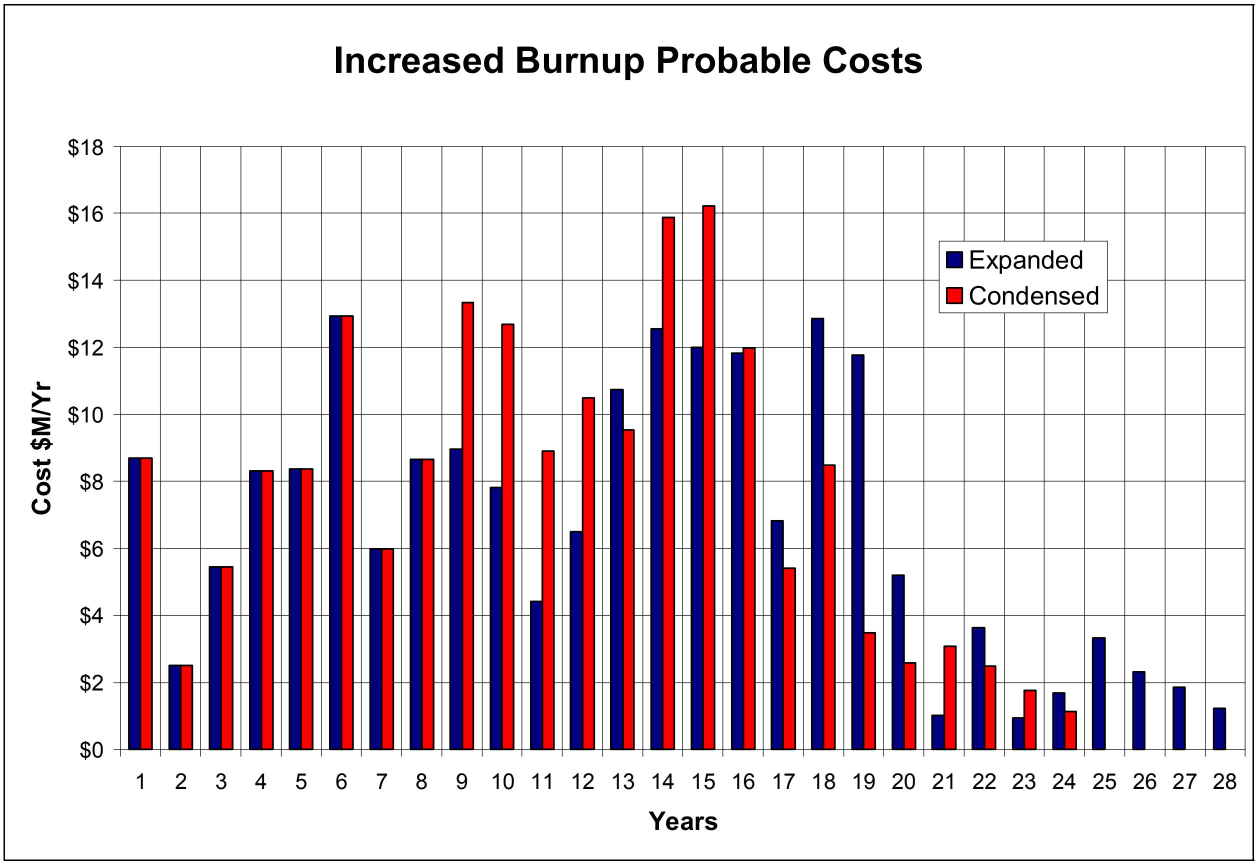

Figure 19 - Comparison of Condensed vs. Expanded increased burnup licensing

From Figure 19 it is easy to see that the trade off for decreased licensing time is an increase in the per year average and the maximum cost in one year. The maximum costs for the Condensed scenario are approximately $4 million more than the maximum cost for the Expanded scenario. This tradeoff does allow for the complete licensing of a full core of 100 GWD/MTU to be completed in 23 years instead of 27. Twenty-three years is approximately 4-5 more years than it takes to license the cladding or fuel alone. The cost of licensing the fuel/cladding through Phase 4 is approximately $190 million dollars or $70 million dollars more than the cost to combine the fuel and cladding licensing for just a full core load. The combined approach that was previously discussed for the fuel/cladding does not license the synergistic effects of the two. However, by licensing the fuel/cladding through Phase 4 the Condensed approach achieves the combined benefit that the fuel and cladding can allow. If this were to mean longer fuel cycles, from 18 months to 24 months, then it could save the licensee $90 million dollars for the final 15 years of operation. 15 years was chosen since it will take ~25years to license this new fuel/cladding combination and most reactors are extending their operating life by 20 years which leaves the majority of plants with 15-20 years of operating life when this new product would be available. This is a savings of billions when evaluated to include all of the current nuclear power plants.

Chapter 5 – Conclusions

By combining various sources from the NRC, to other vendor documents, to experience from engineers who have worked on licensing, overall approaches for licensing new fuels, cladding, and geometry were developed. These specific approaches outlined the common advantages and disadvantages of changing the physical properties or physical structure of the standard fuel assembly. Once the common advantages and disadvantages were determined, the actual steps needed for licensing could be undertaken. First, the models that would need updated were analyzed. From these models, an analysis of the fuel assembly with new fuel, cladding, or geometry could be performed. These models and analyses are the backbone of the rest of the licensing approach. Once the models and analyses are completed, they will be sent to the NRC for approval. Once the NRC approves them, the new fuel, cladding or geometry, can proceed on to the next stage.

The next stage will be to fabricate a lead test assembly that contains test rods (LTATR) for irradiation. These rods will be irradiated for 6-12 months in a research reactor to provide irradiation data that will be added to the models that were previously revised. Once a PIE is performed on the LTATR, the data will be incorporated into the models and the models will be submitted again to the NRC in order to gain approval for implementation of lead test assemblies (LTAs).

When NRC acceptance is gained, the LTAs will then be irradiated inside a commercial reactor. They will be irradiated for up to three cycles. At the end of each cycle, one LTA is removed and the fuel rods contained within are evaluated via a PIE. Once the data is gathered from the PIEs it is again incorporated into the models which are then again sent to the NRC for acceptance. At this stage, the next step is to engage in full core loading of the new fuel, cladding or geometry.

The full core loading is meant to allow for the new fuel, cladding, or geometry to undergo the full scale testing that a full core load allows. At this point, the models and analyses and PIEs from the LTATR and LTAs have provided the NRC with sufficient proof that the fuel, cladding, or geometry is safe to operate inside of a nuclear reactor. The irradiation of a full core load is a major milestone that leads to the most important part of licensing.

Rulemaking is the most important part of licensing. It allows for changes to the existing requirements that govern the requirements of nuclear fuel, cladding and assemblies. By having enough evidence to change the requirements, it eliminates the need to repeatedly request the NRC to allow for changes to the current operating license of a commercial nuclear plant. The drawback is that rulemaking can take years and has the possibility of being rejected. However, if through rulemaking the requirements are able to be changed, then the new fuel, cladding, or geometry has carte blanche to be used in all nuclear reactors. This ability to be used in all nuclear reactors is what allows the maker of nuclear fuels to ensure that their product will be able to be used by all without further testing or licensing.

An increase in the burnup limits to allow for fuels with longer fuel cycles to be used was evaluated. The evaluation of this mirrors the approaches used to gain full approval of the new fuel, cladding, or geometry. It uses LTATRs, LTAs, full core loading and rulemaking to allow for an increase in the burnup of nuclear fuels.

After the licensing approaches for the fuel, cladding and geometry were determined, the cost structure and timeline data was determined from various sources. These sources were other engineers that have done licensing work, various outside documents that give timelines for completion of their work, and timeline data from the NRC. Once the timeline data and costing was gathered it was only a matter of ensuring the enough man power was assigned to each task. The cost structure and timeline data was evaluated for each of the three licensing scenarios, a combination of the fuel and cladding and finally an increased burnup through Phase 4.

Chapter 6 – Recommendations

In order to increase the profitability of the new fuel, cladding or geometry, there are three main physical changes that would increase the profitability of the new fuel. These three physical changes are increased density to allow for more U-235, increased thermal conductivity to increase heat transfer from the fuel, and decreased heat capacity to allow for a lower centerline temperature. A combination of these three allows for longer burnups, higher thermal output, and a larger margin for safety.

The advantages of a new cladding are higher thermal conductivity, lower corrosion rate, minimal insulating oxide layer, higher operating temperatures, and a resistance to the steam-cladding reaction and oxide formation. These changes would allow for a safer fuel that can operate at higher burnups and at higher temperatures.

Finally, the advantages of a new geometry are higher heat transfer areas per volume, higher mixing and lower pressure drops through the core. These would lead to a larger thermal output per volume from the core that would not sacrifice flow rate.

For the timeline and cost structure five different scenarios were analyzed. These scenarios were, fuel, cladding, and geometry by themselves, fuel and cladding both simultaneously and combined, and fuel and cladding combined to phase 4. From the timeline and cost structure analysis, there are some distinct advantages for combining the fuel and cladding licensing into one large project that are not available for doing each one alone. These advantages included a reduction in man power and cost for the combined approach. This reduction in manpower was approximately \$7.6 million over the life of the project. One scenario that should be evaluated for this combined fuel/cladding approach would be to irradiate the LTATR and LTA such that the fuel and cladding costs for the PIE can be combined into one PIE for both. This would save \$20 million for the life of the project and would bring the cost to just over \$100 million over approximately 18 years.

Because of the high cost of the geometry, and because it would need to be combined with both the fuel and the cladding to reach phase 4, the additional ~\$75 million needed to combine all three was determined to be too costly. This was because of the high costs and the large scale changes to the models and analyses that will need to be performed before the geometry licensing could proceed to the LTATR stage. It is recommended here that unless changes to the geometry would lead to larger profit returns than could be realized from the combined fuel/cladding approach then the geometry approach should not be performed. Again this is assuming that there all three are available for licensing and the purpose is to choose between them. If this is not the case, i.e. only geometry is available as an option, as long as the profitability is acceptable to management, the licensing of the geometry should proceed.

Since the combined fuel/cladding approach has the possibility of very large savings, it was chosen to use this scenario as the baseline for analyzing the Phase 4 increased burnup section. The cost of the increased burnup was determined to be almost \$190 million which is spread over either 23 or 27 years depending upon whether the condensed or expanded versions are chosen. If there is an ample supply of funding for the project, the condensed version should be chosen. This will allow for maximum usage of the increased burnup fuel/cladding combination. It is possible for an additional \$20 million savings when using the combined fuel/cladding licensing approach. This brings the total cost down to almost \$170 million but the most important change is that it decreases the maximum amount of funding needed in one year from ~\$16 million to ~\$12-13 million. Because of the larger savings and the possibility for longer fuel cycles it was determined that the combined fuel/cladding increased burnup fuel assembly is the most economical of all. This is because this new fuel will have the possibility to reduce the fuel costs to the licensee and should in turn make this fuel assembly very attractive in comparison to the other scenarios. Also, since the condensed timeline (increased burnup) only increases the timeline by four years over the combined fuel/cladding licensing approach, the advantage of higher burnup should outweigh the increased time and cost.

Bibliography

"Atomic Energy Commission." Encyclopædia Britannica. 2005. Encyclopædia Britannica Premium Service. 10 Oct. 2005 <http://www.britannica.com/eb/article-9010129>.

Code of Federal Regulations Section 10, Appendix A; Amended December 23, 1999; http://www.nrc.gov

Code of Federal Regulations Section 10, Appendix A; Amended December 23, 1999; http://www.nrc.govNIST Webbook, Phase Change Data http://webbook.nist.gov/cgi/cbook.cgi?ID=C1344576&Units=SI&Mask=4#Thermo-Phase

Patent Disclosure: U.S. Patent No. 6,909,765 Lahoda, Edward J., Westinghouse Electric Company LLC, Pittsburgh, PA; June 21st 2005

STD-CP-03-30 – [Proprietary], Task, K.D., November 13, 2003, Westinghouse Science and Technology Division

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ACADEMIC EXPERIENCE:

M.S. in Engineering, Chemical Engineering, West Virginia University, 2005

Thesis: *Evaluate the Adaptability of Present Licensing Requirements to New Nuclear Fuels for Use in Current Reactors*
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B.S. in Chemistry, West Virginia University, Morgantown, WV, 2004

PROFESSIONAL EXPERIENCE:

Intern, Westinghouse Electric Company, May 2005-present

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Chemistry Department, West Virginia University
Course: Organic Chemistry I & II, Freshman Chemistry II

Intern, FirstEnergy Corporation, May 2004-August 2004

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PUBLICATIONS:
Hydrogen Costs for the PBMR Thermal Reactor and the Westinghouse Process, 2005 AICHE Annual Meeting Proceedings; David McLaughlin, Samuel A. Paletta, et.al.