

University of Ontario of Institute of Technology

2000 Simcoe St N,

Oshawa, ON

L1H 7K4

**ENGR-4998U: Thesis Design Project I**

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**Review Design and Material Selection of Fuel Cladding Materials for Possible Further Enhancement**

***Prepared for:***

Dr. Brian Ikeda

***Prepared By:***

Navneet Bhalla

King Chi Kwan

Brian Liang

Samantha Perry

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### Abstract

The increase in demand for power over the last decade has led some countries to consider the more efficient Generation IV Supercritical Water Reactors (SCWR) as a power generation method. The unique operating environment and lack of information make it difficult to properly design such a reactor. This thesis focuses on the fuel cladding component of an SCWR. By performing a literature review on pertinent background information and using a nominal group technique ranking model, a set of selection criteria with pertaining factors were developed. These selection criteria were designed to test the validity of a particular design. Verification of the model will be done during thesis II.

**1.0 Introduction**

In the past decade, the demand for power has increased. Growing populations and increasing industrial inputs have become more taxing on our energy infrastructure. With the possibility of new electrical uses (transportation or home/office uses) and the demand for cleaner and more efficient energy, the demand for power in general is forecasted to increase. A method of meeting these demands would the protection and introduction of a new type of Nuclear Power Plant, the Supercritical Water Reactor (SCWR) which uses supercritical water as the working fluid. Nuclear Power has always been the cheapest to the consumer and with the increases in efficiency and high supply of uranium, SCWR seem like a very real option. However, an SCWR has never been built and people are not yet sure if this design can still be safe given the unique operating conditions. The fuel cladding is one such material that would be affected by the environment, thus it is necessary to study how fuel cladding would need to be designed.

In an ideal world, cladding would not be used at all. From a performance standpoint, the cladding impedes heat transfer and impedes neutron transparency which reduces the overall efficiency of the plant. However, fuel cladding is necessary because it contains the radioactive nuclides of the fuel. When the fuel undergoes fission, fission gasses and other fission fragments are formed. These highly radioactive particles pose a risk to the health and safety of the public and environment and need to be contained. If these particles were to enter into the moderator or PHT system, they would be more difficult to contain. The cladding is the first barrier after the fuel itself to provide a way to contain the radionuclides and easily remove them from the system. Thus it is necessary to have cladding, which in addition to offering better control of flux and containment of fission fragments, also provides and easy method of loading and unloading fuel from the calandria. The choice of material needs to be selected to resist these operating conditions while still providing adequate heat and neutron transfer to the working fluid.

**2.0 Problem Statement**

The current issue is that there is no known fuel cladding design that has been confirmed to be functional in the SCWR’s operating conditions. CNSC requires a means of being able to select and verify a material for fuel cladding which could be utilized in an SCWR.

To address this problem we have created a set of selection criteria for the SCWR fuel cladding which will aid in the selection process and verification of a given fuel cladding design. These criteria were designed around the aspects of performance, function and safety of the SCWR’s fuel cladding. This report will detail the method that used for developing the SCWR fuel cladding design criteria

**2.1 Scope**

The focus of Thesis I is to develop selection criteria for fuel cladding in a conceptual supercritical water reactor. We will be investigating into the background of other systems where the real world operating conditions are similar to those found in a conceptual SCWR. This will be focused mainly on the in-core performance. We will also be performing literature reviews to derive various criteria for which factors will fall under.

We will not be reviewing information on other types of Gen IV reactors as they do not utilize SCW and thus have different operating conditions.  We will not be conducting any experiments or constructing any components, this is beyond the scope of thesis I. We will also not be performing deep analysis on the non-incore factors which affect material selection for Thesis I.

During Thesis II Once a final set of criteria is drafted, the validity of the criteria will be tested by taking a list of potential material candidates and running them through the criteria. If the remaining material candidates are deemed logically sound, then the criteria can be deemed valid.

A secondary screening test will also be made which will take into account the manufacturability, costs, storage/disposal, and other non-operating factors which affect selection.

1. **Background**
   1. **Fuel Cladding**

**3.1.1 Uses and Properties**

Fuel cladding acts as a barrier to radioactive nuclide release, preventing the leaking of materials into the PHT coolant. It is the first line of defense after the pellets themselves and consists of a cylindrical metal sheath which is capped at each end. The cladding holds in the fuel pellets and helium gasses, as well as contains the fission products (this includes the fission fragments and fission gasses) which result from operations. Cladding failure may occur if the proper material is not chosen, or if safe operating parameters are not kept.

The main features of fuel cladding are to contain fuel products, maximize neutron transparency, maximize heat transfer and minimize corrosion. In the case of an SCWR, the operating conditions are harsher than other others as in addition to the high temperature and pressures from supercritical water, there is also irradiation from the fuel.

**3.1.2 Failure Scenarios**

It is important for fuel cladding to be able to withstand failure scenarios but no more than is what is justified by the risk. Thus the cladding material needs to withstand transient effects and smaller scale failures, but should not be designed around extreme failure scenarios. Failure scenarios are caused by a decrease in coolant flow, abnormality in reactor pressure, abnormality in reactivity or LOCA. A total LOCA would result in extremely high temperatures and a high pressure differential which could melt the cladding or cause a rupture. However, it would be unreasonable to expect a material to withstand such temperatures. Given the unlikelihood of such an event and the stringent criteria this would impose, it is not necessary to design for such situations. On the other hand, a smaller LOCA only introduces a smaller temperature increase and thus needs to be designed for.

What would be designed for would be accidents or abnormal events that cause a change in coolant flow. A reduction in the coolant flow would result in greater fuel temperatures which affects fuel cladding. The temperature from a minor accident at the outlet header can be up to 750˚C (Lefu Zhang, 2011).

Another possible scenario could be a change in reactivity. Control rods being removed or liquid control zones being drained. This causes an increase in reactivity which means a greater number of fissions. The increased number of fissions increases the energy released, which increases the temperature.

Shown in Appendix 9.6 the accidental scenarios are briefly discussed.

* 1. **Super-critical Water Reactor**

**3.2.1 Design Consideration**

The design of supercritical water cooled reactor (SCWR) is seen as the natural and ultimate evolution of today’s conventional nuclear reactors. The SCWR design operates above the critical point of water which is at temperature of 3740C and pressure of 22.1 MPa. Table 1lists the range of operating parameters for SCWR. However, there are various SCWR conceptual designs being considered by the different countries. Since, Canadian nuclear safety commission is one of the stakeholders in this thesis project and all the power reactor built in Canada are of the Canada Deuterium Uranium (CANDU) reactor type. Thus, the reference design has been selected for the SCWR system is pressurized – channel type, thermal neutron spectrum and heavy water moderator similar to CANDU reactor. In spite of this, few key feature of reference SCWRs are different from CANDU reactor. Firstly, the light water coolant operating at single phase fluid and properties between those of a liquid and a gas. Secondly, the high operating temperature and pressure will result in the reactor be fuelled offline in batches. Also, the reactor core will be positioned in a vertical orientation. Furthermore, high pressure and temperature inside the reactor core require a new design of the fuel channel. The fuel channel designs for reference SCWR is the high efficiency channel. The High Efficiency fuel Channel (HEC) consists of a pressure tube, a ceramic insulator, a liner tube, and fuel bundles **(**Figure 3). The outer surface of the pressure tube is exposed to a moderator. The purpose of using an insulator is to reduce the operating temperature of the pressure tube and heat losses from the coolant to the moderator. The liner is intended to protect the ceramic insulator from being damaged during operation or possible refueling due to stresses introduced by fuel bundles and from erosion by the coolant flow. In addition, the reference SCWR will have 43 element fuels (CANDFLEX) bundle design similar to that designed for the Advanced CANDU reactor. The CANFLEX bundle consist of two element sizes: Small- diameter element in the outer and intermediate rings and larger diameter elements in the inner and center rings (Figure 4). Special buttons are attached to the element at 2 planes, to provide improved heat transfer. The small – diameter of the outer ring result in a slight larger end-plate diameter compared with end plate diameter of standard 37- element bundle. CANFLEX fuel is designed to have hydraulic and neutronic characteristics that are similar to those of the existing fuel. The fuel bundle, in all other respects, is designed to be equivalent to the 37-element bundle, to be transparent to all reactor system. Table 4 lists the selected parameters of CANFLEX Bundle.

**3.2.2 Nuclear Fuel Options**

High operating temperatures of SCWRs could lead to high fuel centerline temperatures. The existing nuclear power plant utilize uranium dioxide as its fuel; however, it has low thermal conductivity can exceed the industry limit of 1850˚C (Peiman, Pioro, & Gabriel, 2012). Therefore, the potential fuel options for future use in SCWRs are classified into two main categories; metallic fuels and ceramic fuels. The high operating temperature and pressure of SCWR eliminate various most of the metallic fuels options mainly due to their low melting points and high irradiation creep and swelling rates. On the other hand, ceramic have good dimensional and radiation stability and are chemically compatible with most coolants and sheath materials, which make these fuels suitable candidates for SCWR applications. The ceramic fuels currently being examined are UO2, MOX, ThO2, UC, UN, UO2–SiC, UO2–C, and UO2–BeO (Peiman, Pioro, & Gabriel, 2012).

**3.3 Other Reactors and Power Plants**

**3.3.1 Sodium-cooled Fast Breeding Reactors**

Since the concept of SCWR is recently introduce to the nuclear industry, lots of data are physically not available; hence is important to study existing reactor that share similarity in their working condition to assist the selection of cladding materials for SCWR. Although there will be no reactor can fully match all properties of the SCRW, but by making connection with similar working condition reactors and their associated cladding selection, will widen the possible materials options for the SCWR.

The GenIV nuclear reactors promote a better energy efficiency, sustainability and safety; by improving the fuel burn-up rate and reduction of nuclear waste (Azevedo. 2011). Currently there are six Gen IV nuclear reactor concepts; there are SCWR, Sodium-cooled fast reactor (SFR), Lead-cooled reactor (LFR), Gas-cooled fast reactor (GFR), Molten salt reactor (MSR) and Very high temperature reactor (VHTR); within those six design concepts only LFR, SFR, MSR and VHTR have substantial operation experience (World Nuclear Association, 2013). Furthermore, among those four designs the SFR has a comparable reactor core outlet temperature and irradiation embrittlement properties with the SCWR conceptual design (Zhang, Bao, & Tang, 2012).

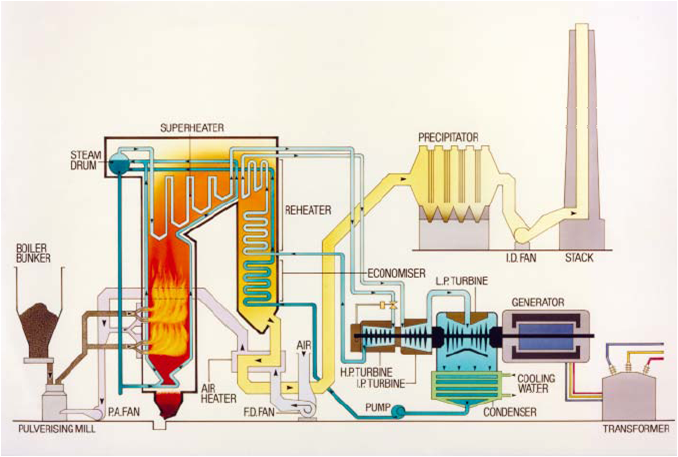
Currently there are constructed experimental SFRs, example are the BN-600 from Russia and Experimental Breeder Reactor II (EBR II) from the United States of America (World Nuclear Association, 2013); able to obtain statistics from a built reactor will further enhance the credibility of cladding material selection process.

The concept of SCWR have neutron spectrum option of fast or thermal, which is dependant to the fuel option; so the Displacement of Atom (dpa) a measurement unit for irradiation can reach up to 150 dpa (Baindur, 2008). As within the SFRs the life time dose are between 150-200 dpa (Los Alamos National Laboratory); which cover the upper bound of the SCWR irradiation dose. Consequently, the irradiation experiences for SFR can also apply to the SCWR scenario (Zhang, Bao, & Tang, 2012).

The cladding material selections for SFRs are Ferritic/ Martensitic Stainless Steel alloy and Oxide Dispersion Strengthened (ODS) alloy (Azevedo. 2011). As those alloys show their capability to handle high operating temperature and irradiation protection; they are enlisted to the potential SCWR fuel cladding material selection list.

**3.3.2 Supercritical Coal-fired Power Plant**

When trying to understand conceptual supercritical water reactors, and the cladding environment, it is useful to find similar operating environments. This makes it possible to understand the materials used in real world, but they need to be put into context for their respective design requirements. Care should be taken in that, the systems with similar operating environments are not identical, and that the design choices were only made to meet the specific requirements. In SCWRs, the conditions are harsher than those found in other plants which means criteria needs to be more stringent to ensure appropriate cladding is selected. A supercritical coal fired power plant is shown:



**Figure 1: Supercritical Coal-fired Power Plant (Appendix 8.1)**

Power plants around the world typically function by utilizing a system which converts heat energy into electrical output. The heat energy is created via a steam generator which in the case for coal-fired plants, involve the feeding of coal into the steam generator which is then heated releasing the binding energy. For sub-critical plants, this would be done by heating water until it changes into steam (Pulverised Coal Combustion with Higher Efficiency, n.d.). The steam passes across and spins a turbine connected to a generator. The axial motion can be used to create electric power. The steam is then condensed back into water and fed back into the system and reheated to undergo the process again. This cycle can be mathematically modeled by the Rankine cycle to determine efficiency.

For a supercritical plant, the process is the same except instead of utilizing water as the working fluid, supercritical water is used instead (Westenhaus, 2008). This is simply water that is beyond the critical point, that is, the water is heated and pressured to a point where it begins to display properties of both a liquid and a gas (Pioro & Duffey, 2007). The operating environment for these steam generators are 22.1 MPa and 3740C.

Efficiency is a function of the difference in the heat transport system and the heat source. Because of the higher pressure and temperature of this working fluid, the temperature difference reduced and there is greater operating efficiency (Westenhaus, 2008). The higher pressure and temperature steam have different thermodynamics including higher expansion pressure and temperature steam (Westenhaus, 2008). This means the turbines are pushed with more force and there are greater yields. Most coal-fired plants operate at 33% efficiency but the conversion to supercritical water can enable 45% efficiency. Typically these a SC plant will operate at 39% efficiency (Westenhaus, 2008).

While the concept for supercritical water reactors is new, supercritical water power plants are not. With regards to coal fired plants, there are two types of plants that utilize these technologies, supercritical water plants and ultra-supercritical water plants (Westenhaus, 2008). Supercritical coal-fired plants were developed back in the 1950s and started seeing implementation in the 1960s (American Electric Power Company, 2013). The first plant was in Ohio at AEP’s Philo Unit 6 (American Electric Power Company, 2013). Ultra-supercritical water plants are the same thing except they operate at pressures higher than the supercritical temperature and sets steam temperatures above 11000C (American Electric Power Company, 2013). Supercritical water has been used throughout many plants around the world and has proven to be more efficient than regular water at converting thermal energy into electrical power.

The issue with supercritical technology is that traditional materials are not able to operate for extended periods of time. A material needs to be able to handle the mechanical stress and thermal conditions. In coal-fired plants, the vessel as well as various components in the steam generator, turbine and piping systems can be made of chrome and nickel-based super alloys. Some of the creep rupture tests showed that Inconel 740 is the best at operating in 7600C, followed by Haynes 320 and CCA 617 (Vis Viswanathan, 2008). In addition, supercritical water is also very corrosive; even more so than regular water.

When it comes to a Supercritical Water Reactor, the same temperature and pressure conditions exist. These were used to create pressure and temperature criteria which would affect fuel cladding. Corrosion criteria could also be created by investigating the corrosive conditions within the steam generator vessel and the supercritical water. Ultra-critical operating conditions should also be noted because they are the next logical step after supercritical water reactors. Finally, it should be noted that the presence of a radiation field in the reactor means that the failure pressure, temperature and corrosion numbers are subject to change. To minimize this, there should be a safety margin developed beyond the maximum operating conditions found in SCW coal-fired plants.

**Potential Cladding Materials Selection**

Candidate of SCWR Fuel Cladding Materials Selection are based on current and history of different power plants, properties of alloys and potential alloy application.

1. Zirconium Alloy   
   Zirconium Alloys are well known fuel cladding material for the pressurized water reactors (PWR) and boiling water reactor (BWR); due to it has a low neutron absorption cross section and effective corrosion resistance. In this design project Zircaloy 2 and Zircaloy 4 the two most common zirconium alloys will be place through the screening process for the cladding material selection. (Zhang, Bao, & Tang, 2012)
2. Austenitic Stainless Steels  
   Austenitic Stainless Steels show a strong potential since it has been successfully operating in the Light Water Reactors (LWR), Fast Bleeding Reactors (FBR) and Supercritical Water-Cooled Coal Fired Plant (SCFP). Some potential candidates are 304 Stainless Steel, AL-6XN, HR3C, SAVE25 and NF 709, those alloys either shown promising result in current power plants or having similar ability mechanical properties. (Zhang, Bao, & Tang, 2012) (Azevedo, 2011)
3. Advance Nickel Alloys (Inconel Alloys)   
   Advance Nickel Alloys show capability to handle high operating temperature and good creep rupture properties within the ultra-supercritical fossil fired plants. Since the SCFP demonstrates a similar working environment, common cladding material like Inconel 740 and Inconel 800HT would be investigate for possible candidate material for SCWR. (Smith & Shoemaker, July) (Azevedo, 2011)
4. Refractory alloys   
   Refractory alloys are manufacture from refractory metals like Niobium (Nb), Molybdenum (Mo), Tantalum (Ta), Tungsten (W) and Rhenium (Re). Refractory metals are elements which have a melting point greater than 2000°C, offering a high melting point for the refractory alloys. Some refractory alloys like Ta-10W, W-Re and Mo-0.5Ti-0.1Zr will be place into the screening process, since it been suggested for fuel cladding of Lead-Cooled Reactor (LCR) and Gas-Cooled Fast Reactor (GFR) to suit their high operation temperature range. (Azevedo, 2011)
5. Oxide Dispersion Strengthened (ODS) alloys   
   Oxide Dispersion Strengthened alloys been through a process of addition of thermally stable Nano scale particles, (Ti2O3 titania and Y2O3 yttria) dispersed into the ferritic matrix, allowing the alloy to have a higher creep strength, better radiation protection and swelling resistance. Alloy MA 956s’ mechanical properties show its capability of handle high temperature and have low corrosion rate; thus it also will place under the screening process. (Azevedo, 2011) (Allen, Burlet, Nabstad, Samara, & Ukai, 2009) (Allen, Burlet, Nabstad, Samara, & Ukai, 2009)
6. Ceramic materials   
   Ceramic Materials like Silicon Carbide (SiC) and Zirconium Carbide are newly introduce to the nuclear industry; their research and development result show a high temperature strength, low corrosion rate and good irradiation behaviour. Their unique development should also put into the screening process to help analyse the possible cladding material for SCWR. (Azevedo, 2011) (Bragg-Sitton, Barrett, Van Rooyen, Hurley, & Khafizov, 2013)

**3.2 Non-Physical Design Criteria**

**3.2.1 Material Costs**

The involved considerations here are the cost of the material. A material needs to be financially reasonable if it is to be used.

Although cost is major consideration for fuel cladding material selection, but the design team focus is on performance, function and safety. Thus cost was not placed as an independent criteria, instead it been placed as a factor under the miscellaneous criteria and will have further investigate in Thesis II.

**3.2.2 Material Manufacturability**

Even if the material passes all the criteria, to be a valid candidate it also needs to be able to be manufactured. A material that cannot be manufactured will be set aside in the candidate list, where further research in the material will be conducted when it becomes possible to manufacture.

1. **Methodology**

**4.1 Physical Design Criteria**

**4.1.1 Developing Design Criteria and Factors**

Design criteria were developed from researching operating parameters of power plants and nuclear power reactors. Reactors with similar operating conditions to the conceptual SCWR were chosen when developing the design criteria factors. By reviewing all the factors that could affect the fuel claddings performance, function and safety, as well as any other factor deemed relevant to fuel cladding, we were able to generate a list of over 50 factors. For example, since the SCWR environment operates at a high temperature and pressure, we were able to develop criteria for strength and temperature that the SCWR must meet. Corrosion of the fuel cladding material is another criteria obtained from literature review and some information from coal-fired plants. We also inferred the existence of irradiation criteria as this is a nuclear powered reactor and thus the fuel itself could have irradiation effects on the cladding. The range for this was developed based on the background from sodium fast breeder reactors which have similar radiative conditions to the conceptual SCWR.

**4.1.2 Rationale for Ranking Important Criteria**

While having over 50 criteria meant a broad coverage of many factors, these needed to be grouped and sorted down into criteria in order for them to give an idea to the degree that it affects the main criteria. The ranking of the important criteria is based on the performance, function and safety of the cladding. The nominal group technique was used in the decision analysis process when determining the important fuel cladding factors. The nominal group technique evaluates important factors by ranking them according to their ability to meet and affect the performance, function and safety requirements (Institute, 2013). The ranking is from 0 to 5. A value of 5 represents the criteria that highly important to fuel cladding and 0 meaning not important to fuel claddings performance, function or safety. The matrix for ranking each of the criteria is shown in 5.1 Design Criteria Matrix. The criteria chosen for the final design criteria will receive a total ranking of 10 or higher. Economics received 0 ranking overall of but will be analyzed. However, cost and the ability to manufacture the material will be crucial to the material chosen but will not influence the performance, function and safety of the fuel cladding. These features can be ranked according to performance, function and safety requirements of cladding. The main performance requirements for fuel cladding are the inability to release radionuclides into coolant and the ability to transfer heat from the fuel to the coolant. The main function requirement for fuel cladding was the ability to contain the fission products. The main safety requirement was the ability to influence failures and be physically stable. For example, activation cross section has an influence on the heat transfer ability and less on the ability to contain fission products. The factors ranked with a score of 10 or higher can then be placed into criteria categories developed through literature review. The main categories will be corrosion, heat transfer, irradiation, strength and miscellaneous. Miscellaneous category will be criteria that do not fit in the other four categories. Further investigation into these 4 categories will be conducted in thesis II by each group member.

**5.0 Results**

**5.1 Design Criteria Matrix**

The design criteria matrix shown in Table 1: Design Criteria Matrix displays the results of the performance, function and safety requirements. Each requirement was ranked with a score between 0 and 5. The factors that were ranked with a total score of 10 or higher was deemed important.

**Table 1: Design Criteria Matrix**

|  |  |  |  |  |
| --- | --- | --- | --- | --- |
| **Cladding Factors** | **Performance** | **Function** | **Safety** | **Total** |
| Activation Cross Section | 5 | 3 | 4 | 12 |
| Activation Energy | 5 | 4 | 4 | 13 |
| Alloy Chemical Composition | 2 | 1 | 2 | 5 |
| Annealing Heat Treatments | 4 | 4 | 5 | 13 |
| Conductivity | 5 | 5 | 5 | 15 |
| Coolant Chemistry | 5 | 3 | 5 | 13 |
| Coolant Temperature | 2 | 2 | 1 | 5 |
| Corrosion Rate | 5 | 5 | 5 | 15 |
| Corrosion Resistant | 5 | 5 | 5 | 15 |
| Cost | 0 | 0 | 0 | 0 |
| Cracking | 5 | 4 | 5 | 14 |
| Creep Rupture Strength | 4 | 5 | 5 | 14 |
| Creep Strength | 4 | 5 | 5 | 14 |
| Density of supercritical water | 3 | 3 | 3 | 9 |
| Ductility | 4 | 4 | 3 | 11 |
| Embrittlement | 4 | 4 | 4 | 12 |
| Enthalpy | 3 | 4 | 3 | 10 |
| Erosion | 5 | 5 | 5 | 15 |
| Exposure Time | 4 | 5 | 4 | 13 |
| Fatigue | 4 | 3 | 3 | 10 |
| Fuel Density | 3 | 2 | 3 | 8 |
| Fission Gas Release | 4 | 3 | 3 | 10 |
| Heat Capacity | 4 | 5 | 4 | 13 |
| Heat Flux | 5 | 5 | 5 | 15 |
| Hermeticity | 3 | 2 | 3 | 8 |
| Hoop Strength | 5 | 5 | 5 | 15 |
| Hydraulic Resistant | 5 | 4 | 5 | 14 |
| Hydraulic Flow | 5 | 4 | 5 | 14 |
| Hydriding | 5 | 5 | 5 | 15 |
| Irradiation | 5 | 5 | 5 | 15 |
| Irradiation Creep Strain | 5 | 4 | 5 | 14 |
| Maximum Operating Temperature | 5 | 5 | 5 | 15 |
| Melting Point | 5 | 5 | 5 | 15 |
| Metallurgical Structure | 4 | 4 | 5 | 13 |
| Minimum Operating Temperature | 3 | 3 | 3 | 9 |
| Neutron Absorption Cross Section | 5 | 5 | 5 | 15 |
| Neutron Capture Cross Section | 5 | 5 | 5 | 15 |
| Neutron Density | 4 | 3 | 3 | 10 |
| Neutron Efficiency | 4 | 4 | 4 | 12 |
| Neutron Production | 3 | 4 | 3 | 10 |
| Neutron Spectrum | 3 | 3 | 4 | 10 |
| Operation Lifetime | 3 | 3 | 3 | 9 |
| Oxidation Resistant | 4 | 4 | 5 | 13 |
| Oxidation Temperature | 3 | 2 | 3 | 8 |
| Pitting | 5 | 4 | 5 | 14 |
| Pressure | 5 | 5 | 5 | 15 |
| Radiation Embrittlement | 4 | 5 | 4 | 13 |
| Radiation Resistant | 4 | 4 | 4 | 12 |
| Specific Heat | 5 | 5 | 5 | 15 |
| Strain | 5 | 4 | 5 | 14 |
| Strength | 5 | 5 | 5 | 15 |
| Stress Corrosion Cracking | 5 | 5 | 5 | 15 |
| Stress Hardening | 4 | 4 | 5 | 13 |
| Surface Finish | 3 | 4 | 3 | 10 |
| Swelling | 4 | 4 | 5 | 13 |
| Tensile Strength | 4 | 4 | 5 | 13 |
| Tensile Stress | 4 | 4 | 5 | 13 |
| Thermal Creep Strain | 4 | 4 | 4 | 12 |
| Thermal Conductivity | 4 | 3 | 4 | 11 |
| Thermal Efficiency | 3 | 4 | 4 | 11 |
| Vibration | 4 | 3 | 5 | 12 |

**5.2** **Final List of Design Criteria**

**Table 2: Final List of Design Criteria**

|  |  |  |
| --- | --- | --- |
| Radiation Criteria | For a fast configuration, must able to withstand radiation levels of 150-200 dpa without significant strain effects  For a thermal configuration, must be able to withstand radiation levels of 10-15 dpa without significant strain effects | |
|  | Activation Cross Section | Kept low to increase transparency |
|  | Transmutation | Must not form dangerous isotopes |
|  | Irradiation Induced Creeping | Must not fail |
|  | Irradiation Induced Swelling | Must not fail |
|  | Irradiation Induced Embrittlement | Must not fail |
|  |  |  |
| Corrosion Criteria | Less than 5%, Not to exceed 25 um over 3-4 fuel cycles | |
|  | IGSCC | Must not fail |
|  | Irradiation assisted SCC | Must not fail |
|  | Corrosion Fatigue | Must not fail |
|  | Uniform Corrosion | Must not fail |
|  | Hydriding | Must not fail |
|  |  |  |
| Thermal Criteria | Must withstand 800 oC | |
|  | Operating Temperature | Withstand operating temperatures for long periods of time |
|  | Thermal Creeping | Must not fail |
|  | Melting | Cannot melt in normal or accident scenarios |
|  | Thermal Conductivity | Should be maximized for energy efficiency |
|  |  |  |
| Strength Criteria | Needs to withstand all pressure and strain effects in normal and accident scenarios | |
|  | Operating Pressure | Must withstand normal and accident scenarios, this is 27.5 MPa |
|  | Hoop Stress | Must withstand 10 MPa (this is the differential across the cladding) |
|  | Yield Strength | Must withstand at 650-800C, 150 or 100 MPa respectively |
|  | Fatigue |  |
|  |  | |
| Miscellaneous Criteria | For further consideration in Thesis II | |
|  | Flow Induced Vibration | Needs more information |
|  | Cost | Needs to be reasonable |
|  | Manufacturability | Can it be machined? |

**5.3 Definition of Design Criteria**

**Radiation**

The activation cross section is a measure of the probability of a nuclear reaction occurring. When an atom is irradiated, it exposes only a portion of the nucleus to the radiation energy or particle. This is the part of the nucleus that if hit with sufficient energy, will “activate” the nucleus and cause the necessary nuclear reaction. This includes both the microscopic and macroscopic cross section, they govern the cross section of a specific nuclei and of specific lump of material (with its density) respectively. This is indicated typically in surface unit’s barns. Other important concepts to understand from this are the neutron cross section, neutron capture cross section, neutron efficiency which all defines the transparency to neutrons that the cladding has.

It is important to have a high transparency (and subsequently a low cross section) to ensure that the cladding does not absorb too many neutrons, allowing efficient heat transfer to the working fluid. Some issues with high neutron absorption are creep effects and transmutation. Creep strength is a measure of material resistance to deformation after undergoing a constant weight or force for a period of time. In this case, the neutrons impinging on the materials lattice structure weakens it. During irradiation, the kinetic energy dismantles the atomic lattice first, and lattice restoring forces then reconstruct it, atom-by-atom. In high-dose irradiation, each atom may be displaced from its lattice site many times, the standard measure of which is the dislocations per atom (Lefu Zhang, 2011). This is similar to the radiation conditions found in a sodium cooled fast breeder reactor, the cladding must be able to handle the 15 dpa without experiencing significant embrittlement, swelling and creep effects that could compromise the cladding (Maloy).

Another issue is transmutation, many properties of an element or material can change in the presence of neutron radiation. For example, cobalt-59 upon absorbing a neutron becomes cobalt-60 which is a strong gamma emitter; and therefore all cobalt alloys are unsuitable for reactors.

The radiation criteria require that the fuel cladding not deform under irradiation. For a fast reactor configuration, this means it needs to must able to withstand radiation levels of 150-200 dpa (dislocations per atom) (Maloy). For a thermal reactor configuration, the radiation levels are at 10-15 dpa (Lefu Zhang, 2011).

**Corrosion**

Corrosion is another aspect that needs to be looked into when considering fuel cladding criteria. In any reactor, the working fluid tends to be water which reacts with the cladding and corrodes. This corrosion is based on the activation energy for the chemical reaction between the cladding and the oxygen or hydrogen, forming oxides and hydrides respectively. In the case of the supercritical water environments, the high temperatures and pressures further disassociates the bonding on the water molecules which means it more readily reacts, lowering the activation energy of the reaction. It is necessary to have a material whose reaction with the environment has a high activation energy (it requires a large amount of energy to form the oxide and the reaction will not happen as readily).

There are many modes by which corrosion could occur and all factors that affect these modes will need to be looked into. Flow accelerated corrosion for example is a function of the flow and surface roughness, a smooth surface finish is necessary to minimize these corrosive effects. Stress corrosion cracking is another issue where the combination of corrosive environment with the high pressures can cause a failure that neither condition alone could have caused. For supercritical water, it was found that cracks and exfoliation of surface oxide films always happens on stainless steel and F/M steels when SCW temperatures of 6000C or greater are met (Lefu Zhang, 2011). At these temperatures, the thinning of the cladding is accelerated. It is necessary to have as criteria that the material must not form cracks or exfoliation when exposed to the SC water operating conditions (Lefu Zhang, 2011). Another important corrosion aspect is hydriding which is the process by which a metal corrodes in the presence of hydrides.

The criteria for corrosion effects are that the corrosion rate is not to exceed 5% in wall thickness during 3-4 fuel cycles. At a 5% wall thickness reduction, the tube wall stresses increase by 10% which greatly reduces the safety margin of the cladding (Lefu Zhang, 2011).

**Strength**

Structural properties are another important factor which forms the basis of design criteria. Structural failures are even more important because of the swelling effects of the fuel and radiation field affecting the interior of the cladding as well as the high pressure and corrosive environments on the outside of the cladding.

The toughness of a material is a function of the ductility, brittleness, hardening treatments (can be done through annealing heat treatments), grain size, etc. Ductility is the measure of a materials tendency to deform when under tensile stress without breaking; brittleness is how easily a material breaks without significant deformation. Hardening is a technique to improve the yield strength before failures occur through a change of the materials lattice structure. These need to be taken into account when analyzing the tensile and compressive stresses on the cladding.

A compressive effect results from the interior swelling and exterior pressures acting in directions which oppose one another and the tensile strength is a result of the end caps on the fuel bundles being stretched during operation. The interior effects from swelling also introduce hoop stresses into the cylindrical cladding; this can cause a deformation of the cladding such as ballooning or buckling collapse. The material must not fail under any of the structural modes; it needs to have values for structural properties such that significant deformation could not happen under normal and emergency conditions. These strains can also exist as a result of thermal or irradiation conditions.

The deformability of cladding from two batches of UO2 gauges was evaluated by using resistance strain gauges to measure the circumferential expansion of the sheath as function of power. The results obtained for this investigation on the effects of start-up rates on fuel expansion and the strain cycle to be expected in a load-following reactor are shown in Figure 5. For the first cycle from zero to full power and back to zero power, they agreed well with each other and with the values calculated from simple physical models. However, while the two batches of U02 were thought to be identical, one seemed to deform plastically above 10000C while the other showed non-plastic behaviour up to the maximum temperature of about 18000 C for the rate of power increase in this experiment. The top graph of Figure 6 indicates the local circumferential strain that occurred at this interface and the predicted value. The sum of this and the strain at the pellet midpoint gives the maximum local strain of the sheath. Figure 5 also shows that the sheath recovers very little on its strain as the power is reduced. During subsequent power cycles the recovery is even less, and after an irradiation of about ten days, a return to zero power causes approximately 0.1 % change in sheath circumference. Such small changes in average sheath strain could partly result from strain localization.

With regards to the buckling, an infinite cylinder model is used for the fuel rods. This is given by the equation:

Where E is Young’s Modulus, t is temperature, and D is diameter. If the operating pressure exceeds the collapsing pressure than the cladding will fail and the material cannot be used. By using a safety factor of 1/3 and applying a 110% of the operating conditions, this takes enters the equation at 27.5 MPa and 8000C. This creates criteria for the material in that it must successfully operate at 27.5 MPa at 8000C. That is, calculated pressure must be less than the collapse pressure.

For the strength criteria, the material needs to be able to withstand all operational stresses without significant deformations that could cause a failure. It also must withstand strain induced from thermal or irradiation conditions.

**Thermal**

Thermal criteria were developed to address the important failure modes for cladding that are related to the heat such as melting, swelling, or cracking. During fission, particles and energy is released which is equivalent to 200 MeV. This energy is absorbed by the coolant and reflected by the moderator to heat up the coolant. This creates an internal source of heat which is partially absorbed by the cladding and is a function of the thermal conductivity. A higher thermal conductivity means a lower heat capacity, greater heat flux and greater specific heat. Under optimal conditions, the thermal conductivity is maximized so fuel transfers the maximum amount of heat to the coolant as possible; this would minimize the temperature effects on the cladding and reduce the likelihood of it experiencing a heat based failure.

In SCWRs, this relationship is more complicated because conductivity also becomes a function of irradiation as a result of change in the chemical and physical composition of the cladding. The major factors are temperature, porosity, oxygen to metal atom ratio, PuO2 content, pellet cracking, and burn up. The second largest resistance to heat conduction in the fuel rod is due to the gap. Therefore, fuel centerline temperature profiles will be calculated based on a no-gap condition and with gap widths of 20 μm and 36 μm (Peiman, Pioro, & Gabriel, 2012). The calculation will also be performed to determine the temperature of the fuel in the radial and axial directions. The sheath temperature is high at SCWR conditions; therefore, it is necessary to take into account the radiative heat transfer.

The hottest location for the cladding would be the interior walls which can have potential temperature of 18500C(the industry maximum for the centerline temperature of the fuel pellets). However, the cladding would never be exposed to such temperatures except in extreme accident scenarios. The outer walls can experience inlet and outlet temperatures which can be in the range of 2700Cto 6750Crespectively in the conceptual design (Pioro & Duffey, 2007).

For the thermal criteria, it is required that the cladding be able to withstand up to 8000C without failing or melting.

**Miscellaneous**

Other important factors include fretting which is the result of the flow induced vibrations which are a result of the physical interactions of the working fluid with the cladding or the erosion effects from flow and debris collection (Yoshiaka Oka, 2010). Embrittlement which is caused by the temperature transients introduced from LOCA, from irradiation, hydrogen corrosion effects, etc. (Yoshiaka Oka, 2010). There also exist cost and manufacturability factors which will be taken into further consideration during our Thesis II.

**6.0 Conclusion**

Although currently there are no operating SCWRs in the world yet, through the process of literature reviews on SCWR conceptual designs, SFR and SCW coal-fired plant; the design team was able to develop a set of selection criteria for fuel cladding. By utilizing the given SCWR estimates and other comparable power plants, we could generate a set a design criteria to assist the selection of fuel cladding material for the SCWR. The design criteria were defined by using a nominal group technique ranking model; the decision model was based around the aspects of performance, function and safety. Upon completion of the ranking of all factors, five major criteria were for formed through literature review: Thermal, Corrosion, Strength, Radiation and Miscellaneous. The narrowed list of factors was sorted down into the criteria as they pertained most relevant towards. The criteria and important factors are will be more thoroughly examined in Thesis II, where each member will evaluate the validity of the criteria as well as add in or retract factors as deemed necessary. Furthermore, a list of potential cladding materials was drafted and more may be considered in Thesis II.

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**8.0 Appendix**

**8.1 Table 3: Range of Operating Parameters in SCWR (Pioro & Duffey, 2007)**

|  |  |  |
| --- | --- | --- |
| **Parameter** | **Unit** | **SCWR CANDU** |
| Power Thermal | MW | 1350 - 2540 |
| Power electrical | MW | 515 - 1220 |
| Thermal eff. | % | 38 - 48 |
| Pressure | Mpa | 24 - 26 |
| Tin Coolant | oC | 270 - 400 |
| Tout Coolant | oC | 380 - 625 |
| Core Height | m | 3.2 - 6 |
| Enrichment | % wt. | 3.5 - 6 |
| Tmax Cladding | oC | <850 |
| Fuel centerline temperature | oC | <1850 |
| Average burn – up | MW\*Days/Kg | approx. 45 |
| Maximum linear heat rate | KW/m | <39 |
| Flow Dynamic design consideration | MPa | <0.02 |
| Pressure difference |  | <1/3 buckling collapse pressure |
| Heated fuel channel length | m | 5.772 |
| Compressive to yield strength ratio | % | <0.2 |
| Cumulative damage fraction | % | <1 |

**8.2 Abnormal and Accidental Events**

**Abnormal Event**

* Decrease in Core coolant flow
* Partial loss of reactor coolant flow
* Loss of offsite power
* Abnormality in reactor pressure
* Loss of turbine load
* Isolation of main stream line
* Pressure Control system Failure
* Abnormality in reactivity
* Loss of feedwater heating
* Reactor coolant flow control system failure
* Uncontrolled CR withdrawal at normal operation
* Uncontrolled CR withdrawal at start-up

**Accidents**

* Decrease in core coolant flow
* Total loss of reactor coolant flow
* Reactor coolant pump seizer
* Abnormality in reactivity
* CR ejection at full power
* CR ejection at hot standby
* Change of Liquid Control Zone Levels
* LOCA
* Large LOCA
* Small LOCA

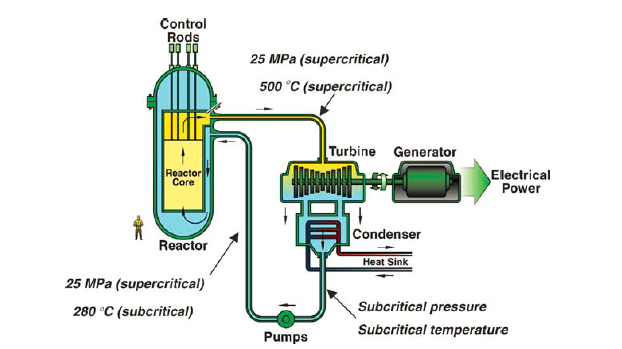
**8.3 Table 4: Proposed SCWR Fuel Cladding & Bundle (Pioro &Duffey, 2007)**

|  |  |  |
| --- | --- | --- |
| **Bundle** | | |
| **# of Fuel channel** | 300 | |
| **# fuel bundle in fuel channel** | 12 | |
| **Number of Elements per bundle** | 43 | |
| **No of element per Ring** | Centre | 1 |
| Inner | 7 |
| Intermediate | 14 |
| Outer | 21 |
| **Outer diameter of intermediate and outer element** | 11.5 mm | |
| **Outer diameter of inner and centre element** | 13.5 mm | |
| **Length of Bundle** | 6 m | |
| **# of heated rods** | 43 | |
| **DHY of fuel channel** | 7.52 mm | |
| **DH of fuel channel** | 9.04 mm | |
| **Heated area of fuel channel** | 9.26 m2 | |
| **Flow are of fuel channel** | 3625 mm2 | |
| **Pressure tube inner diameter** | 103.45 mm | |

**8.4 Table 5: Full List of Design Criteria**

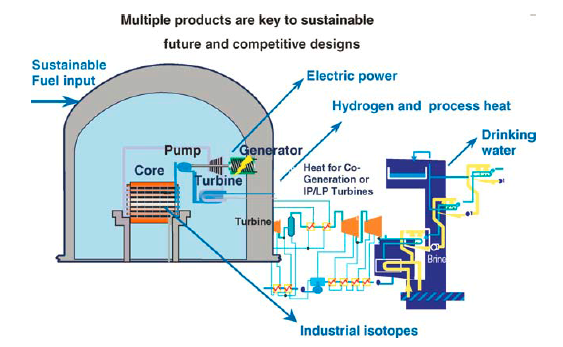
|  |  |
| --- | --- |
| **Cladding Factors** | Irradation |
| Activation Cross Section | Irradation Creep Strain |
| Activation Energy | Maximum Operating Temperature |
| Alloy Chemical Composition | Melting Point |
| Annealing Heat Treatments | Metallurgical Structure |
| Conductivity | Minimum Operating Temperature |
| Coolant Chemistry | Neutron Absorption Cross Section |
| Coolant Temperature | Neutron Capture Cross Section |
| Corrosion Rate | Neutron Density |
| Corrosion Resistant | Neutron Efficiency |
| Cost | Neutron Production |
| Cracking | Neutron Spectrum |
| Creep Rupture Strength | Operation Lifetime |
| Creep Strength | Oxidation Resistant |
| Density | Oxidation Temperature |
| Ductility | Pitting |
| Embrittlement | Pressure |
| Enthalpy | Radiation Embrittlement |
| Erosion | Radiation Resistant |
| Exposure Time | Specific Heat |
| Fatigue | Strain |
| Fuel Density | Strength |
| Fission Gas Release | Stress Corrosion Cracking |
| Force | Stress Hardening |
| Heat Capacity | Surface Finish |
| Heat Flux | Swelling |
| Hermeticity | Tensile Strength |
| Hoop Strength | Tensile Stress |
| Hydraulic Resistant | Thermal Creep Strain |
| Hydraulic Flow | Thermal Conductivity |
| Hydriding | Thermal Efficiency |

**8.5 Schematic of US Pressurized-vessel SCW nuclear Reactor**



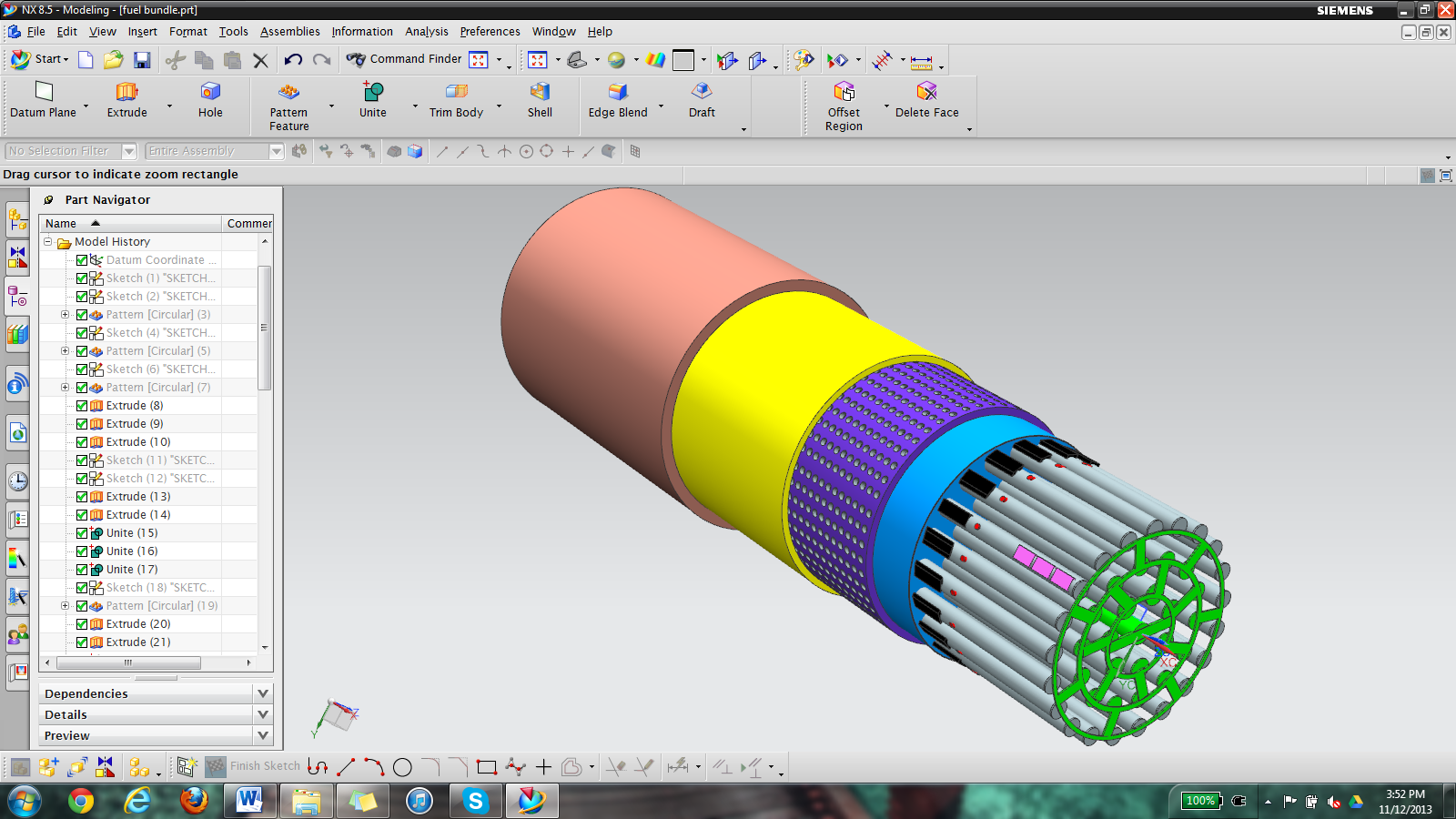
**Figure 2: Schematic of US pressurized-vessel SCW nuclear reactor (Pioro & Duffey, 2007)**

**8.6 Schematic of Pressure Tube SCWR**



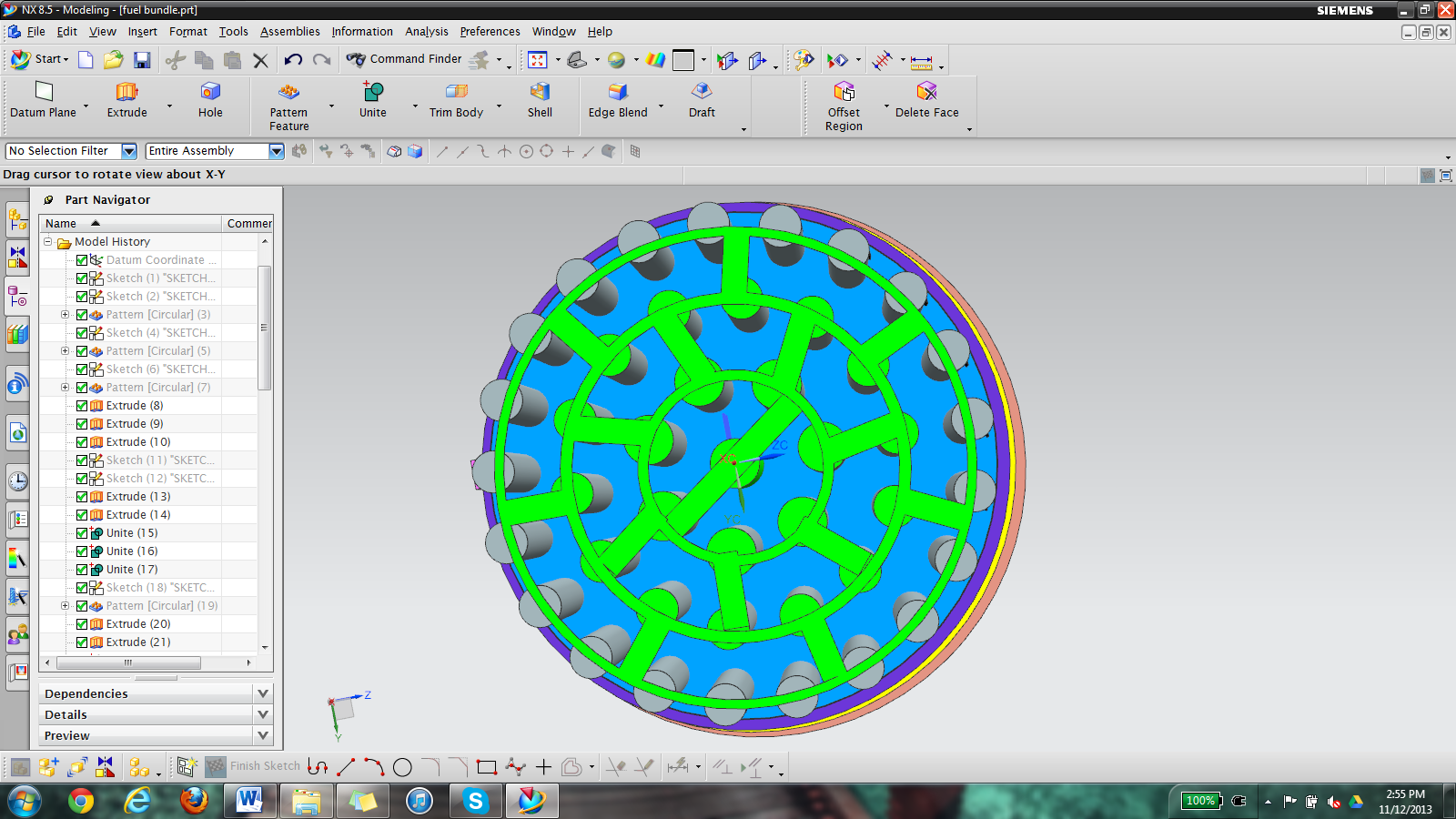
**Figure 3: Schematic of Pressurized Tube SCWR with Hydrogen Production**

**8.7 Figure 4: NX model of SCWR Fuel Bundle**



**Figure 4 shows the fuel bundle in the SCWR. Red shows the pressure tube. Yellow shows the insulator. Purple shows the liner. Blue shows the coolant.**

**8.8 Figure 5: NX model of SCWR Fuel Bundle End Diagram**

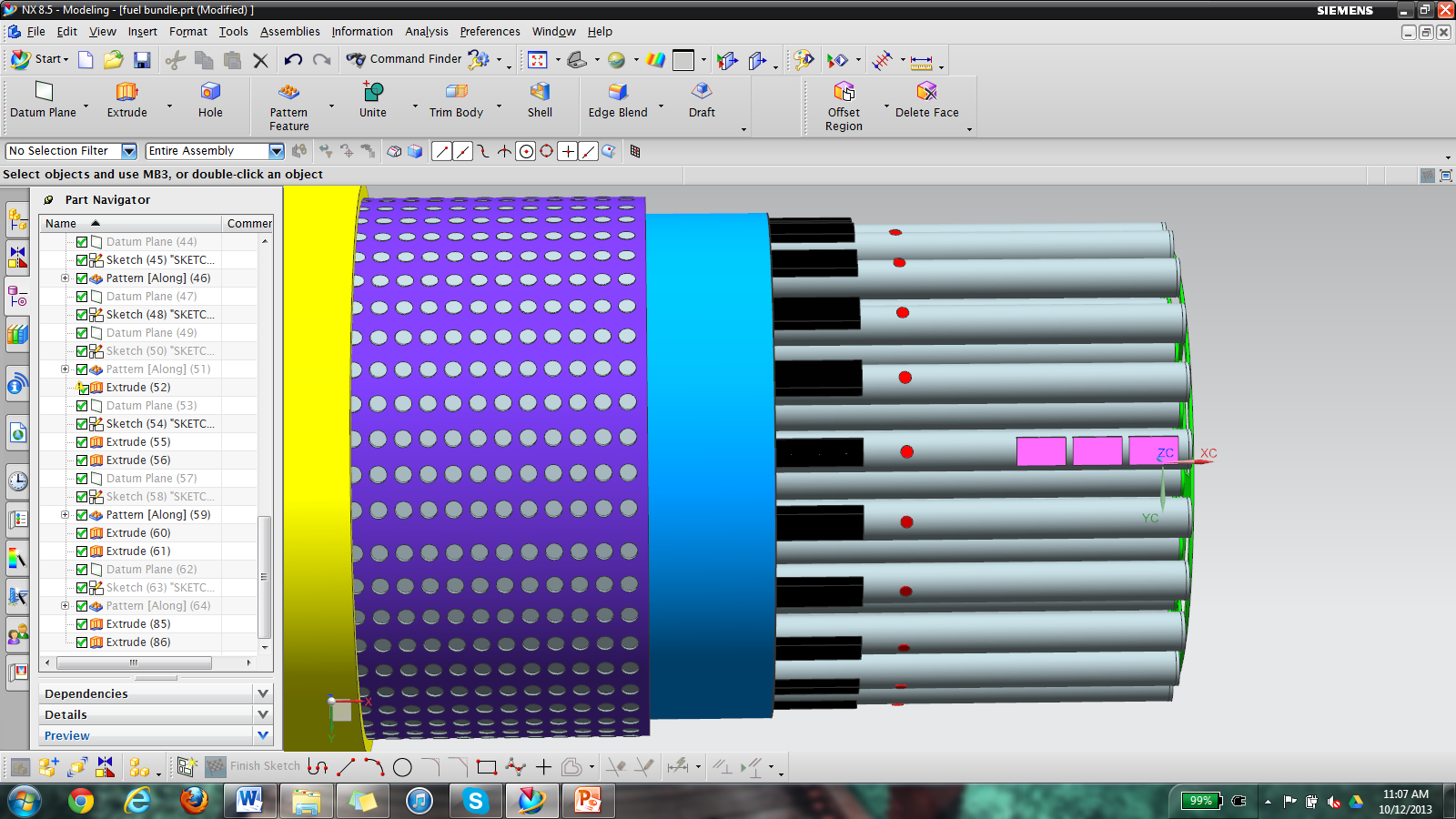


End Support Plate

**Figure 5**

**8.9 Figure 6: NX model of SCWR Fuel Bundle Side Diagram**

Bearing Pads

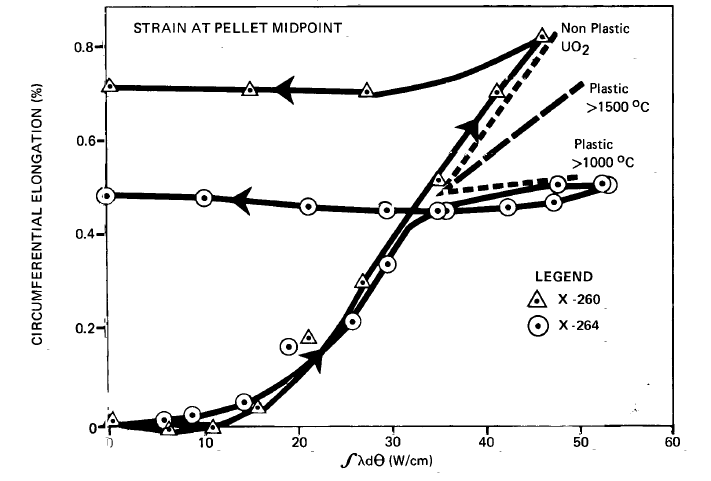


Buttons

Uranium Dioxide Pellets

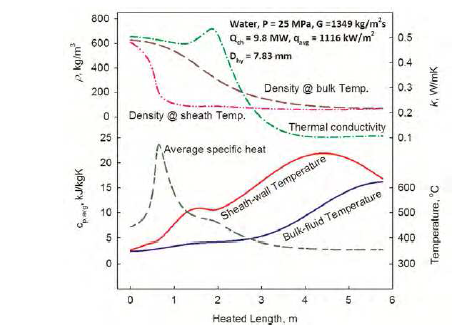
**Figure 6**

**8.10 Figure 7: Pellet Interface Graph**



**Figure 7 Pellet Interface circumferential strains measured resistance strain gauge during the first power cycle (two different test) compared with calculated Expansion**

**8.11 Figure 8:**



**Figure 8**

**8.12 Navneet Bhalla’s Thesis II Proposal**

**8.13 King Kwan’s Thesis II Proposal**

**8.14 Brian Liang’s Thesis II Proposal**

**8.15 Samantha Perry’s Thesis II Proposal**