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Preliminary Neutronic Assessment for ATF (Accident Tolerant Fuel) based on Iron Alloy

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ABSTRACT

After Fukushima Daiichi nuclear accident in 2011, the nuclear fuel performance under accident condition became a very important issue and currently different research and development program are in progress toward to reliability and withstand under accident condition. These initiatives are known as ATF (Accident Tolerant Fuel) R&D program, which many countries with different research institutes, fuel vendors and others are nowadays involved. Accident Tolerant Fuel (ATF) can be defined as enhanced fuel which can tolerate loss of active cooling system capability for a considerably longer time period and the fuel/cladding system can be maintained without significant degradation and can also improve the fuel performance during normal operations and transients, as well as design-basis accident (DBA) and beyond design-basis (BDBA) accident. Different materials have being proposed as fuel cladding candidates considering thermo-mechanical properties and lower reaction kinetic with steam and slower hydrogen production. The aim of this work is to perform a neutronic assessment for several cladding candidates based on iron alloy considering a standard PWR fuel rod (fuel pellet and dimension). The purpose of the assessment is to address different parameters that might contribute for possible neutronic reactivity gain in order to overcome the penalty due to increase of neutron absorption in the cladding materials. All the neutronic assessment is performed using MCNP, Monte Carlo code.

1. INTRODUCTION

Enhancing accident tolerant fuel became a new agenda in the nuclear fuel research and development as consequence of Fukushima Daiichi nuclear accident and the U.S. Department of Energy Office of Nuclear Energy (DOE-NE) initiated an Accident Tolerant Fuel (ATF) development program, within the Fuel Cycle Research.

The main objective and goal of ATF development program is to identify alternative fuel system technologies which can enhance the safety, maintaining or improving the fuel performance during normal operations, operational transients and accident conditions without reducing competitiveness and economics of commercial nuclear power generation.

A several research and development are on going at present days related to ATF in many research labs, universities all around the world (United States, France, Japan, Sweden, Belgium, Russia Federation, etc) and some fuel vendors (AREVA, Westinghouse, General Eletric). Specifically in USA, the US DoE is already providing substantial support for R&D on accident-tolerant fuel concepts with a challenge target to deliver a lead test assembly

(LTA) in an LWR by 2022 [1]. According to general definition, an accident tolerant fuel shall have the following characteristics and attributes: improved reaction kinetics with steam, slower hydrogen production, improved cladding thermo-mechanical properties, improved fuel thermo-mechanical properties, reduced fission products release, stability against irradiation. The nuclear fuel with all desirable attributes can enhance accident tolerance under design basis accident (DBA) and beyond design basis accident (BDBA). Some of attributes are directly related to fuel material properties and others are related to fuel cladding material properties and combination of fuel and cladding materials.

The fuel material shall exhibit enhanced properties as fission products retention, high thermal conductivity (higher than UO_2), consequently will operate at lower temperature; heat capacity smaller than UO_2 that energy deposited will be smaller in case of reactivity insertion accident, such as control rod injection.

Currently, the fuel material under investigation are mainly: FCM (Fully Ceramic Micro-encapsulated Fuel) developed for high temperature gas-cooled reactors; metallic fuel (UMo) which exhibited a very high thermal conductivity (ten times higher than UO_2); Uranium mononitride (UN), which posses a desirable combination of high melting temperature and good thermal conductivity properties; U_3Si_2 fuel has a higher density and high thermal conductivity compared to UO_2 and low parasitic neutron absorption.

The overall conducting researches has not presented any conclusive data yet, FCM fuel need higher enrichment level in order to maintain same cycle length, consequently the economic cost will have a significant impact, Uranium Mono-nitride (UN) requires a nitrogen enrichment facility, other fuels were not proven under irradiation and others drawbacks are not fully addressed[2,3,4].

The cladding material research is more promising for improvement considering existing industrial technology, feasibility and economic point of view. The nuclear fuel industries have been conducting research and from economic development for cladding material since very beginning and can take benefit from almost four decades of activities. At earlier stage of commercial nuclear power generation fuel cladding was iron based alloys (austenitic steels) which reliable and successfully operated over years in many nuclear power reactor but the main limitation was associated to high neutron absorption penalties. At beginning of 50´s, the US Navy propelled zirconium alloy research resulting in current cladding material for majority of LWR commercial nuclear power reactor. After approximately six decades of continuous research and development zirconium alloy exhibits a significant improvement and, all manufactures have been improved the fabrication process, Q&A (quality and Assurance) program and outcome of the all industries effort is a very reliable cladding. Although some limitations and concern of zirconium alloy are well known and challenged, specially under design basis accident condition (i.e. LOCA), it was secondary before Fukushima accident. The cladding material to be investigated as ATF candidate must fulfill some requirement in order to successfully substitute current zirconium alloy cladding, The main requirement are: corrosion resistance, oxidation kinetic, hydrogen pick-up, dimensional stability, enhance performance under irradiation, neutronic performance, compatible with existing LWR thermal hydraulics, compatible with existing fuel transportation cask, ease fuel storage without significant modifications, material availability, manufacturability, not significant costly and licensibility (regulatory/license process) issue.

The actual cladding technologies are being considered and under investigation: Advanced Zr based alloys, Zirconium Alloy with coating and sleeve (Coating as Ti3AlC2, Ti2AlC, Nb2AlC, TiAlN), Ceramic material (SiC) and fiber/SiC matrix, iron based alloy, advanced ferritic/martensitic steel (FeCrAl), refractory material (molybdenum alloy) and innovative alloys with dopants (e.g. chromia, SiC powder, etc)[4,5,6,7].

This paper will focus on neutronic assessment of promising fuel cladding candidates comparing with zirconium alloy clad as reference. The reactivity will be a parameter to evaluate the neutronic performance of each fuel cladding. Moreover some tradeoff analysis will be addressed considering enrichment level, moderation degree, fuel pellet dimension and cladding thickness as parameters.

2. ZIRCONIUM AND IRON-BASED ALLOYS CLADDING OVERVIEW

Totality of commercial LWR fuel cladding is made from zirconium-based alloys, such monopoly of zirconium over almost fifty year is based on performance, reliability, accumulated industrial experience and continuous evolution. Mainly, due to a combination of desirable properties: reasonable corrosion resistance, small neutron capture cross section, good thermo-mechanical properties and metallurgical manufacturability. The zirconium alloy fuel cladding contain 97 up to 99% zirconium and some other minor elements are added to optimize the desired properties, e.g., Sn, Fe, Cr, Nb and Ni. Most of added elements contribute to performance of a cladding alloy, e.g., creep, growth, corrosion and hydrogen pickup. The main limitation of Zr-based cladding is usually determined by its corrosion properties, i.e., oxidation in the hot reactor coolant, and in particular the associated hydrogen pickup in zirconium, which can reduce the mechanical strength and ductility.

The zirconium alloys corrode relatively rapidly in steam environment with high temperatures which always occurs at LOCA (Loss of Coolant Accident)[7]. Such corrosion process combined with hydrogen production is a well know phenomena in safety analysis scenario, moreover the Fukushima accident clearly demonstrated a weakness of zirconium alloy cladding under DBA accident condition. At end, zirconium alloy can deteriorate during the fuel disposition process due to presence of hydrogen inside cladding as zirconium hydride.

The stainless steel as fuel cladding material has a large amount of accumulated experience over almost twenty years of operational experience. The steady state and under transient condition, the performance of PWR using stainless steel clad has been generally satisfactory and no noticeable failures were reported. A total of approximately 600,000 fuel rods had been irradiated up to 1981. Majority of commercial PWR´s were using AISI 304 (stainless steel type) as cladding material also few others using: AISI 316, AISI 304L, AISI 347 and AISI 348 (with improved strength and stress corrosion resistance – cold worked and annealed) types. The good performance presented specially under transient condition indicate higher thermal mechanical margins compared to zirconium alloy, less susceptible to damage due to PCMI (pellet Clad Mechanical Interaction) effect, moreover the stainless steel is resistant to stress corrosion cracking generated by fission products in the fuel.

During DBA scenarios, e.q. LOCA (Loss Of Coolant Accidents), austenitic stainless steel exhibits a metal-vapor reaction rate, an amount of hydrogen production, a reaction rate are lower than zircaloy and the oxygen embrittlement is almost inexistent, consequently, it is expected a smaller cladding deformation (ballooning) and reduced cooling channel blockage. Moreover, in reference [8] mentions comparisons between stainless steel and zircaloy rods under LOCA condition predicted a significant lower probability of rod rupture when using stainless steel. The main disadvantage of iron based alloy is directly related to neutronic performance due to very high neutron absorption cross section (approximately fifteen times when compared to zirconium alloy), which implies a fuel enrichment penalty.

An extensive study was conducted at EPRI[8] to determine the advantages and disadvantages of stainless steel cladding compared to zircaloy, based on the available technology and current economic aspect. Other important study reexamines iron based alloys for their potential application as nuclear fuel cladding to replace zirconium alloys[8]. A several mechanical properties (yield strength, creep rupture strength and Young´s moduli) of iron based alloy for unirradiated and irradiated condition are presented, discussed and compared to zircaloy. The study show a good performance of iron based alloy even under accident condition.

Recently, the authors (Abe, A. & Giovedi, C.) of this work had conducted a comparative fuel performance studies considering stainless steel (AISI-348) and zircaloy using a modified version o FRAPCON[9] code. Essentially the conducted study aimed to modify the exiting FRAPCON fuel performance code in order to perform fuel performance analysis with AISI-348 stainless steel as cladding material. The main outcome of the analysis were a good performance of stainless steel compared to zirconium alloy, specially dimensional changes due to irradiation, no gap closure was observed while zirconium alloy exhibit gap closure.

Nowadays, steels industries have been conducting development and research of advanced steel, tailoring specific properties which will allow thinner walls mitigating the neutronic penalty and enhancing mechanical strength, corrosion resistance and embrittlement.

All improvements will be attractive for ATF international framework, as can be seen by research conducted at General Electric, Westinghouse, Universities and Labs worldwide.[10]

3. PREMILINARY NEUTRONIC ASSESSMENT

This paper will focus on five iron based cladding candidates and compare their neutronic performance with zirconium alloy as reference case. The Table 1 shows a selected iron based alloys and respective elemental composition. The preliminary neutronic assessment will be conducted performing a single unit cell calculation using the MCNP, Monte Carlo code [11]. The neutronic parameter to be evaluated is infinite neutron multiplication factor and the reactivity, which gives the information regarding neutron absorption contribution on cladding material. The fuel depletion condition will be not addressed in this work at moment. The single unit cell calculation considers a standard PWR fuel, with 4.2% of enrichment level and following characteristic data:

Fuel pitch: 1.25984 (cm)
Fuel pellet diameter: 0.819150 (cm)
Fuel clad inner diameter: 0.83566 (cm)
Fuel clad outer diameter: 0.94996 (cm)
Clad thickness: 0.05715 (cm)

The initial reactivity assessment gives a neutron absorption penalty in pcm unit compared to zirconium alloy case. The MCNP code is a general-purpose Monte Carlo code used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code has general 3D geometry modeling capabilities and uses detailed point-wise cross-section data for all physics interactions. Over 836 neutron interaction tables are available for approximately hundred different isotopes and elements.

Initially, the unit cells are modeled in the MCNP code to obtain the infinite neutron multiplication factors (K_{inf}) for the reference case and the other iron based cladding fuel rods using same condition and fuel data (temperature, pitch, diameters, enrichment degree). The boundary condition adopted was reflecting surface for outside surface and enough number of neutron cycles to obtain a reduced standard deviation (~0.0004).

Table 1: Cladding alloys data

Element (wt%)	Zircaloy-4	AISI-304	AISI-348	APMT	Fe20Cr	Fe20Cr20Ni
Zr	98.256	----	----	0.10	----	----
Fe	0.22	71.446	60.646	69.491	80.285	57.916
Ni	----	8.27	11	0.12	----	19.9
Cr	0.11	18.8	17.7	21.6	19.7	20.2
Al	----	0.01	4.9	4.9	----	0.03
Mo	----	0.27	2.8	2.8	----	----
Sn	1.27	----	----	----	----	----
Si	0.01	0.42	0.39	0.53	0.01	0.22
C	0.016	0.028	0.04	0.03	0.002	0.001
S	----	----	----	----	----	----
Hf	----	---	----	0.16	----	----
Y	----	----	----	0.12	----	----
O	0.118	0.006	-----	0.049	0.003	0.003
Mn	----	0.73	1.7	0.1	----	1.61
^{93}Nb	----	----	0.8	----	----	----
^{181}Ta	----	----	0.004	----	----	----
^{59}Co	----	----	0.02	----	----	----
^{139}La	----	----	----	----	----	0.12

3.1 Infinite neutron multiplication factor and reactivity penalty for different alloys

The Table 2 present the results (K_{inf}) obtained from MCNP, Monte Carlo code considering infinite unit cell calculation taking into account 5500 cycles and 40000 neutrons per cycle, which gives an uncertainties around the 0.0004. The reactivity penalty is defined as:

$\Delta\rho=[K_{inf}(\text{reference}) - K_{inf}(\text{Fe-alloy})]*1.00E+05$, and although is not a standard definition in this paper the difference will be an unit of pcm.

Table 2: Infinite neutron multiplication factor and reactivity penalties for different alloys

Alloy	K_{inf}	$\Delta\rho$ – Penalty(pcm)
Zircaloy	1.36204 ± 0.00004	---------
AISI-304	1.24078 ± 0.00004	12,126
AISI-348	1.23651 ± 0.00004	12,553
APMT	1.24134 ± 0.00004	12,070
Fe20Cr	1.25147 ± 0.00004	11,057
Fe20Cr20Ni	1.22576 ± 0.00004	13,628

The results obtained show notably penalties due to neutron absorption, basically due to presence of Fe, Cr and Ni nuclides, which has neutron thermal absorption cross section 2.53, 3.1 and 4.6 barns, respectively. The zirconium neutron absorption cross section is approximately fifteen times lower then iron. Additionally, their content in the alloy composition is very representative, specially Fe (iron) in all alloy, Ni (nickel) in the AISI-304 (SS-304) and AISI-348 (SS-348). The presence of Cr (Chromium) and Ni (Nickel) in alloy gives high resistance to oxidation under high temperature steam environment (e.g. LOCA), but the penalty is roughly 12,000 pcms. The ferritic alloy APMT (Advanced Powder Metallurgic) and SS-304 exhibit almost the same penalty; the contribution of nickel is quite evident comparing Fe20Cr and Fe20Cr20Ni, which gives almost 2,500 pcms of additional loss of reactivity. The AISI-304 and AISI-348 shown almost same penalty.
Although this work has not addressed depletion condition and reactor core calculation with others components (guide tube, spacer grids, burnable poison, etc.), according to reference [12] the expected behavior of penalty under depletion condition could be reduced due to the neutron spectrum hardening. The neutron spectrum hardening is a consequence of fuel depletion; as fuel depleted the plutonium inventory is increased representing overall reactivity gain. Therefore, the reactivity penalty observed at beginning of cycle can be reduced significantly at end of cycle.
To overcome such penalty at beginning of cycle, different approaches can be envisaged: increase of uranium enrichment level, changing the moderation ratio (water channel), reduce cladding thickness, increase fuel pellet diameter, combination of previous mentioned parameters and others.
The first approach considered in this work is a uranium enrichment level changing, the Table 3 shows an increase of enrichment needed to overcame the penalty at beginning of cycle.

Table 3: Increase of enrichment level to overcame neutron absorption alloys

Alloy	Increase of Uranium Enrichment Level
AISI-304	8.0 %
AISI-348	8.0 %
APMT	8.0 %
Fe20Cr	7.5 %
Fe20Cr20Ni	8.5 %

The increase of uranium enrichment level shall have other impacts associated to the fuel cycle activities, starting from fuel fabrication facility (review on criticality safety), storage and transportation of fresh fuel, transportation and storage of irradiated fuel, whereas a new licensing requirement for enrichment above 5.0% would be required for all fuel cycle facilities.
The Figure 1 shows reactivity as function of enrichment level needed to compensate the loss of reactivity due to neutron absorption, the enrichment level span over 4.2% up to 8.5% for all different alloys. However, the increase of enrichment level shall have impact at fuel fabrication cost.

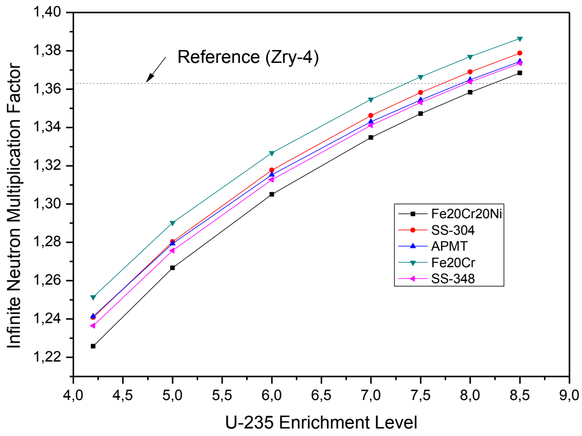

Figure 1. K_{inf} as function of enrichment level (dot-line is reference case)

The required increase of enrichment level is nearly double compared to reference level (4.2% of U-235) at beginning of cycle. The Fe20Cr alloy requires less increase of enrichment level and the on opposite side, is the Fe20Cr20Ni alloy. Nevertheless, the assessment indicates potential economic impact on the fuel cycle due to increase of enrichment.
As neutron moderation plays an important key role in the LWR reactors, another possible strategy to overcome the penalty without changing enrichment level could be a moderation ratio change. The neutron moderation essentially is associated to water channel available in the fuel assembly geometry, the parameter was addressed changing the fuel rod pitch. Initially, moderation ratios span over 20% (plus and minus) from original value for each one of cladding alloy.
The Table 4 shows an influence of moderation in the neutron infinite multiplication factor (K_{inf}), the over-moderate (increased fuel pitch) system present a large reactivity and under-moderate system (reduced fuel pitch) present a reduction of reactivity. Moreover, the similar effect can obtained varying the enrichment level due to spectrum hardening related to presence of the U-238 amount. Although, the work address an influence of the moderation ratio at beginning of life only, the moderation ratio is associated to an important parameter for safety (reactivity coefficient) and, shall be evaluated under depletion condition. The moderator reactivity coefficient must keep negative during entire fuel cycle length.

Table 4: Infinity neutron multiplication factor as function of moderation for iron based alloys

Alloy	Infinity Neutron Multiplication Factor (K_{inf})*				
	-20%	-10%	reference	+10%	+20%
AISI-304	1.01519	1.15582	1.24078	1.29140	1.32077
AISI-348	1.00673	1.14969	1.23651	1.28845	1.31863
APMT	1.00734	1.15221	1.24134	1.29546	1.32723
Fe20Cr	1.20487	1.16609	1.25147	1.30223	1.33155
Fe20Cr20Ni	1.00321	1.14177	1.22576	1.27581	1.30493

- *Standard deviation is ± 0.00004

The increase of moderation ratio changes contribute up to 3050 pcms for iron based alloy compared to zirconium alloy cladding at reference pitch. The higher increase was observed in the Fe20Cr alloy and smaller in the Fe20Cr20Ni alloy. The gain of reactivity is quite significantly compared to reference case of each correspondent alloys, roughly 8000 pcms can be obtained and will remain almost another 4000 pcms of penalty compared to reference case (zirconium alloy).
Moreover, considering only in term of reactivity gain, the increase of moderation ratio can goes up to optimum moderation ratio. The Figure 2 shows the reactivity as function of moderation ratio (percent of reference value) and trend of reactivity shall reach a plateau as pitch increase.
The main outcome of this assessment is a possible reactivity gain due to increase of moderation ratio can compensate part of the neutron absorption penalty and certainly will have less economic impact when compared to enrichment increase.
The pitch size change of 20% could impact directly the number of fuel rods in the actual LWR fuel assembly array (17x17) and possible fuel cycle length. Nevertheless, additional investigation must be conduct in order to evaluate the thermo-hydraulic performance, structural mechanic requirement, full core neutronic performance, economic issues and all safety parameters associated to fuel assembly design.

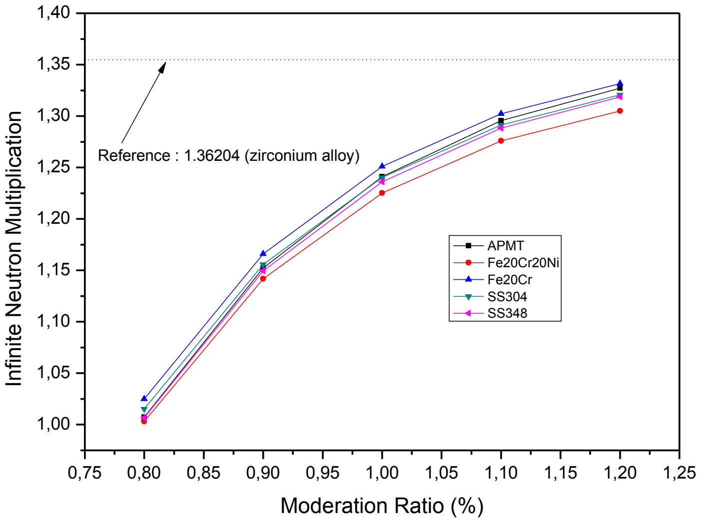

Figure 2. Infinite neutron multiplication factor as function pitch size (dot-line is reference case)

The initial reactivity evaluation shown a main contribution for neutron absorption penalty is due to the presence of Ni and/or Fe in the alloys. In order to address such contribution, the thickness of cladding is also evaluated in this work. The thickness of cladding is reduced up to 20% from nominal value, moreover the thickness change is done in the internal dimension (fuel clad inner diameter) in order to preserve same moderation ratio; otherwise the system could be affected due to changes in two parameters: cladding thickness and moderation ratio.
The cladding thickness reduction can contribute roughly up to additional 2000 pcms in terms of reactivity gain. Some contribution of the neutron absorption penalty can be reduced by means of cladding thickness reduction. As iron based alloy exhibit better mechanical properties compared to zirconium alloy, the reduction could not affect the mechanical performance. Moreover, additional investigation is presented in Section 3.2 conducted using modified version of FRAPCON code[9].

Table 5: Cladding thickness reduction effect

Alloy	K_{inf}*	K_{inf} (reference)+	Reduction of penalty (pcms)
AISI-304	1.26444	1.24078	2,366
AISI-348	1.26081	1.23651	2,430
APMT	1.26494	1.24134	2,360
Fe20Cr	1.27339	1.25147	2,192
Fe20Cr20Ni	1.25173	1.22576	2,597

- *Standard deviation is ± 0.00004
- + reference value is a nominal clad thickness (0.05715 cm)

The cladding thickness reduction strategy allows a proportional increase in the fuel pellet diameter preserving same gap size. The final investigation was done increasing the fuel diameter pellet with conjunction of cladding thickness reduction preserving the original size of fuel rod gap. The new fuel pellet radius will increase about 3% compared to original radius. The increase of fuel pellet radius will represent an overall increase of fuel mass in the reactor core about 6.0%.

The results obtained showed a gain of reactivity is approximately 2,000 pcms at beginning of life for considered cladding alloys comparing with their reference case.

Table 6: Cladding thickness reduction and fuel pellet diameter increase effect

Alloy	K_{inf}*	K_{inf} (reference)+	Reduction of penalty (pcms)
AISI-304	1.26098	1.24078	2,020
AISI-348	1.25730	1.23651	2,079
APMT	1.26109	1.24134	1,975
Fe20Cr	1.26950	1.25147	1,803
Fe20Cr20Ni	1.24895	1.22576	2,319

- *Standard deviation is ± 0.00004
- + reference is a nominal clad thickness (0.05715 cm)

The major outcome of this evaluation of cladding thickness and fuel pellet diameter is an importance of moderation ratio expressed as U/H ratio; although the diameter of fuel pellet was increased and thickness of cladding was reduced, the results showed no reactivity gain at all. Actually, there is a loss of reactivity for each one of the alloys, it is a clear indication of moderation ratio importance. Comparing Table 5 and Table 6 results as presented in Table 7, it can be seen there is no gain of reactivity.

Table 7. Penalties due to cladding thickness and fuel pellet diameter changes

Alloy	Penalty reduction due to cladding thickness (pcms)	Penalty reduction due to cladding thickness and increase of fuel pellet diameter (pcms)
AISI-304	2,366	2,020
AISI-348	2,430	2,079
APMT	2,360	1,975
Fe20Cr	2,192	1,803
Fe20Cr20Ni	2,597	2,319

3.2. Fuel Performance

Additionally to the neutronic assessment, some very preliminary fuel performance was evaluated using modified version of FRAPCON code, named IPEN/CNEN/SS [9]. The FRAPCON-3.4, fuel performance code [13] was developed at PNNL and sponsored by U.S.NRC (the United States Nuclear Regulatory Commission) for the licensing of nuclear power plants. The code originally was developed to evaluate zirconium-based alloys cladding, but was modified to evaluate stainless steel as cladding. Essentially, the modifications were done in the material properties data available at MATPRO material data library and some subroutines associated to cladding properties.

Two cases were selected to investigate the fuel performance under irradiation condition in order to verify the cladding hoop stress, internal pressure and fuel pellet centerline temperature evolution. The Table 8 shows the main data utilized for IPEN/CNEN-SS code simulation; other data required for fuel performance simulation were kept same for both cases and are similar to actual LWR.

The IPEN/CNEN-SS code data required are: initial fuel rod pressure, fuel pellet height, dish size, chamfer size, fuel pellet density, enrichment degree, clad and fuel roughness, fuel rod stack length, fuel rod void fraction, characteristic of fuel plenum spring, plenum height, equivalent channel, axial power profile, number of time steps, length of each time step, etc. The simulation time length is approximately 30,000 MWd/tU of burnup level under steady state condition.

Table 8: Some fuel rod data considered in the IPEN/CNEN-SS code

Parameters	**Case 1**	**Case 2**
Fuel pellet diameter (cm)	0.819	0.843
Fuel clad outer diameter (cm)	0.950	0.950
Fuel clad inner diameter (cm)	0.836	0.860
Fuel gap size (mm)	0.166	0.166
Clad thickness (mm)	0.570	0.450
Enrichment (% U235)	4.20	4.20

The Figures 3, 4 and 5 present the internal fuel rod pressure, cladding hoop stress and fuel centerline temperature evolution, respectively.

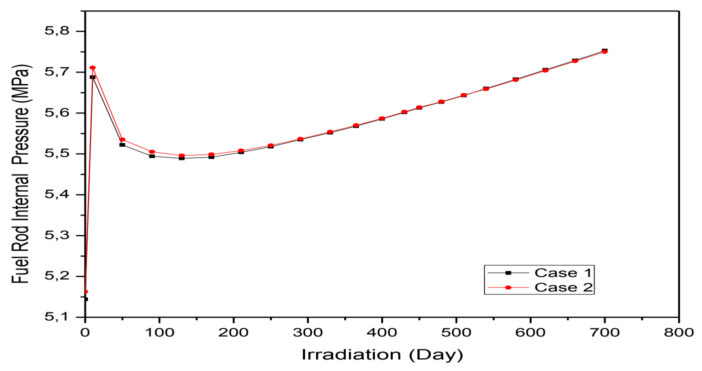

Figure 3. Fuel Rod Internal Pressure Evolution

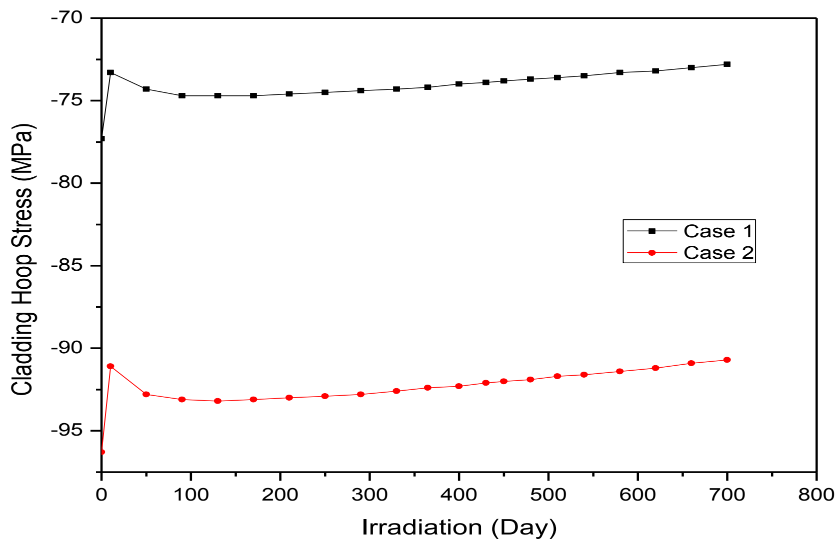

Figure 4. Fuel Cladding Hoop Stress Evolution

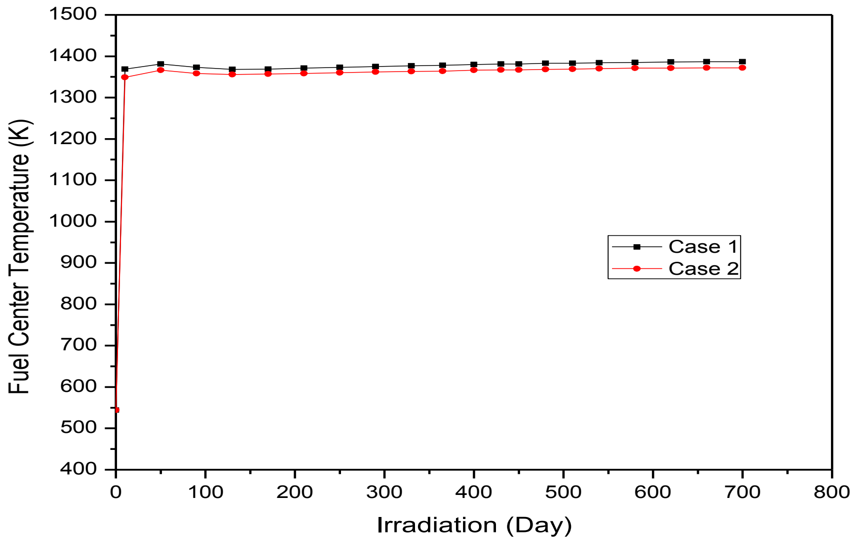

Figure 5. Fuel Centerline Temperature Evolution

According to Figure 3, the fuel rod internal pressure evolution does not show any significant difference between two cases, the initial decrease of pressure is linked to densification of fuel pellet at very beginning of irradiation. Although, the case 2 has larger fuel pellet diameter the overall trend of internal pressure are quite similar compared to case 1(reference case). The additional mass of UO_2 due to increase of fuel pellet diameter does not represent a significant impact at end of irradiation cycle. The Figure 4 shows a clad hoop stress evolution for both studied cases is negative due to the compressive stress imposed by the primary coolant external pressure, which is higher than the internal pressure. The behavior is kept during all the irradiation time due to the open gap. The cladding hoop stress for case 1 is higher due mainly to clad thickness, the internal pressure is nearly comparable and primary coolant pressure is same for both cases and, the small difference observed in the internal pressure is a consequence of the somewhat higher fuel centerline temperature.
The fuel centerline temperature shows the same trend and no significant difference is observed (see Figure 5).
The slightly high temperature for case 1 (more thick clad) compared case 2 is associated to lower heat transfer capacity due to thick cladding.

4. CONCLUSIONS

This work presented some preliminary neutronic assessment of most promising ATF fuel cladding candidates based on iron alloy. The reactivity penalty associated to new the cladding was addressed taking into account enrichment level, moderation ratio, a clad thickness and fuel pellet diameter as variable parameters in order to overcome such penalty. The penalty was quantified in term of reactivity obtained from infinity unit cell calculation using MCNP, Monte Carlo Code, for beginning of cycle. Nevertheless, it is important to consider the contribution of fuel depletion in such quantification.

The assessment results show a possible approach to overcome the neutron absorption penalty due to presence of Fe (Iron) and Ni (Nickel). Moreover, some future activities to be conducted were identified in order to have a better understanding and define a best solution.

The approach taking into account enrichment degree changing shown an increase by approximately double, certainly is the most restrictive constraint from economic point of view. The consequence could be a significant change in the whole fuel cycle. Moreover, economic studies shall be evaluated in order to quantify adequately the real impact.

The moderation ratio evaluation gave an indication of the contribution and importance of moderation phenomena in the LWR reactors. The fuel pitch can contribute significantly to overcome the neutron absorption, but it might affect the geometrical size of fuel assembly, number of fuel rods, thermal-hydraulic parameters and safety. All possible impact associated to pitch change must be investigated properly.

The most efficient strategy to be applied was identified as combination of cladding thickness reduction and moderation ratio change. Moreover, some degree of enrichment increase could be exploited in conjunction with moderation ratio and cladding thickness.

Such approach could drive the best solution for neutronic penalty without introducing significant changes in the fuel technology. Additionally, some very preliminary fuel performance analysis was conducted and overall trend shown a very promising results.

Nowadays, considering the improvement of steel fabrication technology, the cladding thickness reduction will not compromise the structural properties.

Finally, the ATF fuel design must not meet actual performance but, must exceed and should not require substantial modification of existing fuel fabrication facilities. The challenge of ATF comprises many different areas in the nuclear research and development.

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