PRELIMINARY NEUTRONIC ASSESSMENT OF IRON BASED ALLOY FUEL CLADDING

Alfredo Abe[1], Thiago Carluccio[2,3], Pamela Piovezan[2], Cláudia Giovedi[3] and Marcelo R. Martins[3]

[1]Instituto de Pesquisas Energéticas e Nucleares (IPEN / CNEN - SP), Brazil
[2]Centro Tecnológico da Marinha em São Paulo, Brazil
[3]Laboratório de Análise, Avaliação e Gerenciamento de Risco (POLI / USP - SP), Brazil

Abstract. Nowadays two important nuclear fuel performance requirements have been addressed: high burnup in order to improve fuel cycle economic aspect and accident tolerant fuel to enhance the safety under accident condition. The accident tolerant fuel particularly becomes very important issue after Fukushima Daiichi nuclear accident in 2011. The initiatives of R&D program toward to accident tolerant fuel comprises different countries, organizations and including fuel vendors. The 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 been proposed as fuel cladding candidates considering thermo-mechanical properties and lower reaction kinetic with steam and slower hydrogen production, besides that an evaluation of the neutronic aspects for several cladding candidates is important and shall be evaluated. Depending of the outcome of this evaluation, the fuel enrichment level changes to higher than actual level shall be necessary to overcome the neutron absorption penalty. The aim of this work is to perform a preliminary neutronic assessment of fuel cladding based on iron alloy considering a standard PWR fuel rod (fuel pellet and dimension). The main purpose of the assessment is to quantify the penalty due to increase of neutron absorption in the cladding materials and some others fuel parameters are evaluated in order to overcome such penalty. In addition to neutronic assessment, the criticality safety aspects due to increase of fuel enrichment level are briefly presented and discussed.

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 Electric). 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 possess 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 other 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 reactors 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 fulfil 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 Ti_3AlC_2, Ti_2AlC, Nb_2AlC, 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 trade-off analysis will be addressed considering enrichment level, moderation degree, fuel pellet dimension and cladding thickness as parameters.

2. ZIRCONIUM AND IRO-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 contains 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.g. 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 re-examines 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 shows 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 existing 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 was a good performance of stainless steel compared to zirconium alloy, especially 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. PRELIMINARY NEUTRONIC ASSESSMENT

This paper will focus on five iron based cladding candidates and compare their neutronic performance with zirconium alloy as reference case. Table 1 shows a selected iron-based alloys and respective elemental composition.

TABLE 1. CLADDIGN ALOYS 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

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 modelling 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 modelled 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).

3.1 Infinite Neutron Multiplication Factor and Reactivity Penalty for Different Alloys

Table 2 presents 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 uncertainty around the 0.0004. The reactivity penalty is defined as:

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

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.

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

Although this work has not addressed depletion condition and reactor core calculation with other components (guide tube, spacer grids, burnable poison, etc.), according to reference [12] the expected behaviour 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, Table 3 shows an increase of enrichment needed to overcome the penalty at beginning of cycle.

TABLE 3. INCREASE OF ENRICHMENT LEVEL TO OVERCOME NEUTRON ASUMPTION

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.

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.

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.

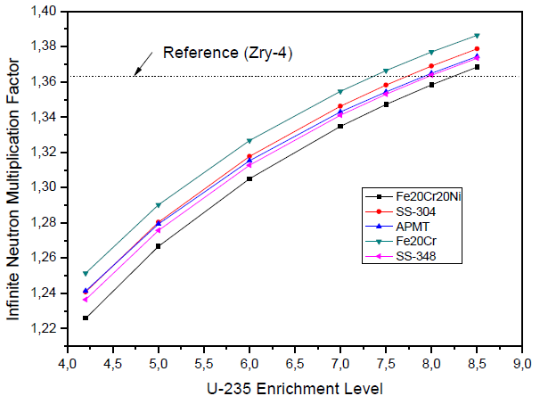

Fig.1 k_{inf} as a function of enrichment level (dot-line is the reference case)

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 be obtained varying the enrichment level due to spectrum hardening related to presence of the U-238 amount. Although the work addresses 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. INFINITE NEUTRON MULTIPLICATION FACTOR AS A 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 go up to optimum moderation ratio. 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.

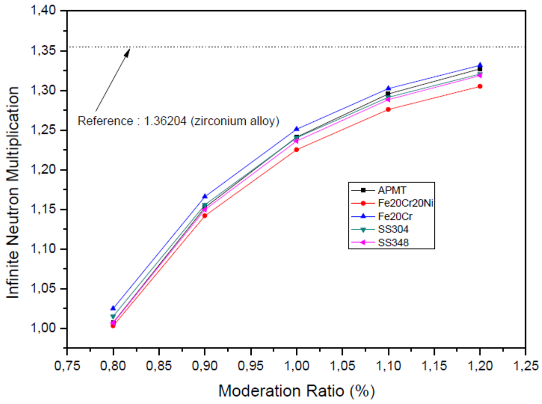

Fig.2 Infinite neutron multiplication factor as a function of pitch size (dot-line is the reference case)

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.

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.

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.

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. 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.

3.2 Criticality Safety Aspect Considering Higher Fuel Enrichment

The neutronic evaluation considering fuel pellet diameter, fuel cladding thickness, moderation ratio shown a clearly evidence of increase of enrichment level will be somewhat required. The changes will certainly affect many design features, safety and operational aspects of actual fuel fabrication facilities around of the world.

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)

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)

Starting from enrichment facility (centrifuge or diffusion process), reconversion and fuel fabrication facilities (powder and pellet), fuel transportation and burned fuel deposition shall be evaluated from criticality safety aspect, in addition to the new license process.

TABLE 7. PENALITIES 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

The criticality safety evaluation considers existing and applicable standards [13-22], qualified methodology, qualified computational tools and nuclear data. Normally, the criticality safety analysis currently conducted to fuel cycle facility is a quite well-known process.
Firstly, the analysis shall start at design phase due to increase 0 enrichment level some equipment/machinery (autoclave, furnace, milling, etc), recipient (tank, vessel, cask, etc), fissile material storage layout, shall be modified properly.

At the design phase some initial calculations are conducted considering reference fissile medium with appropriate an enrichment level. The reference fissile medium is bounding condition which leads to lower limits in order to prevent any criticality risk and keep under sub-critical condition.

The calculations are required to demonstrate that the proposed equipment/recipient/layout/configuration meets the specified safety acceptance criteria.

After design phase evaluation, the analysis shall focus the process conducted at facility, where all relevant activities and steps which fissile material is manipulated shall be identified. This analysis ends imposing some operating control constraints, if necessary, in order to keep under sub-critical condition. The criticality control mode shall be properly implemented taking to account each process and possible operational failure condition shall be analyzed. The criticality safety analysis considers not only normal operation condition, but any malfunctioning, component failure, possible operation anomaly, maintenance activity, etc.

At end, criticality safety analysis report shall be elaborated in order to submit for a license process. Furthermore, operator training under new condition and environment shall be considered at proper time.
Considering actual fuel fabrication facilities following item are identified as important to be considered with enrichment beyond the 5%:

- Enriched UF_6 standard cylinders [23]: Cylinder model 30B, 48Y, has U-235 isotopic content limit to 5% and 4.5%, respectively. A new cylinder shall be developed as industry standard for enriched UF_6.

- Equipment: The size (dimension), capacity (mass and volume) of equipment shall be reduced or implemented a neutron absorber material (cadmium sheet, boron, etc) in all steps related to fuel fabrication.
- Recipient: vessel, tank and others shall be modified, mostly the dimension will be reduced or incorporated a neutron absorber material (cadmium sheet, boron, etc) when is possible.
- Control limit: due to reduced amount of fissile material, production control parameters (batch size) shall be modified properly for each process.
- Storage layout: due to increase of neutron interaction among recipient and/or equipment a new layout shall be implemented and depending of amount of fissile material the facility area can increase.

All modification and changes necessary due to increase of enrichment level shall have economic impact and scale production capacity can be also affected.

Moreover, not directly related to criticality safety of fuel fabrication facilities, transport cask of fresh fuels assembly shall be modified, existing fuel storage pool in the reactor site and the burned fuel transport cask shall be evaluated to criticality safety analysis. Moreover, same critical experiment shall be needed to perform verification and validation steps for currently utilized reactor physics (criticality safety) codes.

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

Finally, substantial modification of existing fuel fabrication facilities shall be taken into account to meet criticality safety requirement due to increase of enrichment degree.

REFERENCES

1. Development Strategy for Advanced LWR Fuels with Enhanced Accident Tolerance, Website on the Internet: Http://energy.gov/sites/prod/files/061212%20Goldner%20-%20NEAC%20Presentation%20%28FINAL%29.pdf, (2012).
2. Pint, B. A., K. Terrani, M. Brady, T. Cheng, J. Keiser; "High Temperature Oxidation of Fuel Cladding Candidate Materials in Steam–Hydrogen Environments," *Journal of Nuclear Materials*, **440**, No.1, pg. 420–427, (September 2013).
3. Shannon B. S.; "Development of Advanced Accident Tolerant Fuels for Commercial LWRs", *Nuclear News*, pg. 83-91, (March 2014).
4. Kristine B., Shannon B. S., Daniel G.; "Advanced LWR Nuclear Fuel Cladding System Development Trade-off Study", INL/EXT-12-27090, (September 2012).
5. Gilles Y. R., Sonat S.; "Enhanced Accident Tolerant Fuels for LWRs – A Preliminary Systems Analysis", INL/EXT-13-30211, (September 2013).
6. Jon C., Frank G., Shannon M. B. S. and Snead L. L.; "Overview of the U.S. DOE Accident Tolerant Fuel Development Program", Proceeding of Top-

Fuel 2013, Charlotte, North Carolina-USA, (September 15-19, 2013).
7. Cathcart, J.V. et al.; "Zirconium Metal–Water Oxidation Kinetics, IV: Reaction Rate Studies", ORNL/NUREG-17, Oak Ridge National Laboratory
8. A. Strasser et al, "An evaluation of Stainless Steel Cladding for Use in Current Design LWRs", ERPI-NP-2642 (1982).
9. A. Abe, C. Giovedi, D. S. Gomes, A. Teixeira e Silva; "Revisiting Stainless Steel as PWR Fuel Rod Cladding After Fukushima Daiichi Accident", *Journal of Energy and Power Engineering*, **8**, pg. 973-980 (2014).
10. K.A. Terrani, S.J. Zinkle, L.L. Snead; "Advanced Oxidation-Resistant Iron-Based Alloys for LWR Fuel Cladding", *Journal of Nuclear Materials* **448**, pg. 420-435 (2014).
11. MCNP: A General Monte Carlo N-Particle Transport Code, Version 5, LA-UR-03-1987, 2003, (Revised 2/1/2008).
12. N. M. George, K. A. Terrani, J. J. Powers; "Neutronic Analysis of Candidate Accident-Tolerant Iron Alloy Cladding Concepts", ORNL/TM-2013/121, (March 2013).
13. ANSI/ANS-8.1-1983; R1988: Nuclear Criticality Safety in Operations with Fissionable Materials Outside.
14. ANSI/ANS-8.1-1998: Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.
15. ANSI/ANS-8.5-1986: Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile.
16. ANSI/ANS-8.7-1975*;* R1982; R1987: Guide for Nuclear Criticality Safety in the Storage of Fissile Materials.
17. ANSI/ANS-8.14-2004: Use of Soluble Neutron Absorbers in Nuclear Facilities Outside Reactors.
18. ANSI/ANS-8.17-1984; R1997: Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
19. ANSI/ANS-8.19-1984; R1989: Administrative Practices for Nuclear Criticality Safety.
20. ANSI/ANS-8.20-1991; R1999: Nuclear Criticality Safety Training.
21. ANSI/ANS-8.21-1995; R2001: Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors.
22. ANSI/ANS-8.22-1997: Nuclear Criticality Safety Based on Limiting and Controlling Moderators.
23. Uranium Hexafluoride – A Manual of Good Handling Practices, ORO-651 (Rev.6), DOE-Oak Ridge.