Silicon Carbide TRIPLEX™ Fuel Clad for Accident Resistance and Durability

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ABSTRACT

The behavior of zircaloy cladding during the loss of coolant accidents at Three Mile Island (TMI) in the US and the Fukushima reactors in Japan was the major cause of overheating and hydrogen release. An alternate cladding material, consisting of multiple layers of silicon carbide fiber matrix composite and silicon carbide high density monolith, (called SiC TRIPLEX cladding) is being developed in the US that offers reliable service during normal reactor operation, and a very large reduction in the exothermic reaction and hydrogen generation that occurred in these accidents. Should another loss of coolant accident occur in a power reactor somewhere in the world, as it surely will someday, use of this cladding would avoid the severe core damage and allow recovery of the plant without total core destruction as occurred at Fukushima and TMI. The status of development and testing of this TRIPLEX cladding is described, as are the plans for future development and testing in preparation for licensing and use. A separate program for replacement of zircaloy channel boxes in BWRs with a unique form of SiC-SiC composite, is also described.

KEYWORDS

Fuel cladding, hydrogen generation. loss of coolant accidents, silicon carbide, BWR channel boxes

1. Introduction

Based on the evidence from both Fukushima and TMI-2, the key to achieving accident resistance in commercial nuclear fuel is the fuel cladding, which is intended to contain the fuel and the fission products as the first line of defense against release of fission products to the environment during accidents. This fission product containment capability of the cladding was lost in both accidents because (1) zircaloy lost all of its strength upon heating above 500 °C and ballooned, blocking flow to the core interior, (2) zircaloy reacted with water releasing large quantities of flammable hydrogen gas, and (3) zircaloy reacted exothermically with water during the accident releasing a large amount of heat. As long as zircaloy is used as a cladding material, with its inherent poor properties above normal operating temperatures, it will continue to be the root cause of fission product release during loss of coolant or other core overheating accidents. For Boiling Water Reactors, an additional root cause of the excessive heat and hydrogen release during the accident at Fukushima was the zircaloy channel boxes which comprise about half the volume of zircaloy in the core, and hence released about half the heat and hydrogen during the hours following the accidental core uncovery.

An alternative to zircaloy that does not react exothermically with water, or release hydrogen when quenched with water at high temperatures, is the ceramic material silicon carbide. Over several decades of research for fusion reactor materials in Japan and the US, it was learned that some forms of silicon carbide are radiation resistant and hence may be usable in fission reactors. However, some research in the last ten years to evaluate use of silicon carbide composites as a replacement for zircaloy in water reactors has proved unsuccessful. For example, an advanced form of SiC-SiC fiber matrix composite fabricated in Japan was tested in the MIT research reactor at normal operating temperatures and found to have excessive corrosion in PWR coolant. Other forms of SiC-SiC had acceptable behavior. A careful evaluation of these experiments has led to a unique form of multilayered SiC that overcomes these difficulties and offers the prospects of high durability during normal operation, and a high degree of stability and fuel containment during severe accidents. This paper describes the unique features in this SiC TRIPLEX™ cladding, including the results of recent

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testing during both normal and accident conditions. It also summarizes the development work that remains as needed to prove the technical and commercial viability of this new fuel cladding.

2. Summary of SiC TRIPLEX Cladding and SiC Channel Box Technology

The CTP TRIPLEX™ cladding is a three layer, all silicon carbide, tube that has the same dimensions as zircaloy cladding in current LWRs. It retains its strength to temperatures above 1400 ^{o}C, has at least 800 times slower reaction rate than does zircaloy at design basis accident conditions, and reduces the amount of hydrogen released during such accidents by at least a factor of 400. It is illustrated in Figure 1 below.

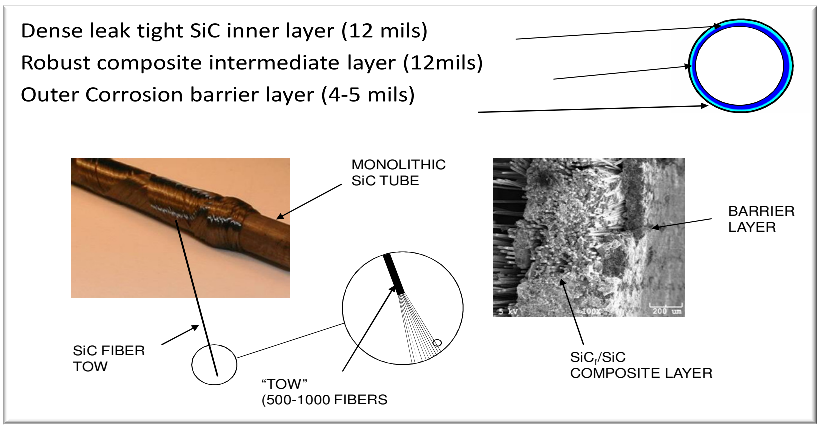

Figure 1 – TRIPLEX™ Fuel Clad Construction

Each of the three layers fulfills a different design requirement. The inner high density monolith layer of stoichiometric beta phase SiC assures hermeticity and fission gas retention during normal operation and reactor transients. The central composite layer made from stoichiometric beta phase SiC fibers and an SiC matrix produced by the Chemical Vapor Infiltration (CVI) process, assures a graceful failure mode even when the clad is subjected to high external forces during accidents, thus avoiding the brittle failure mode of monolithic ceramics. This central layer assures that the clad tube retains its shape and solid fission product retention capability during severe accidents. Alternative matrix infiltration processes (other than CVI) have been tried and proved unsuccessful during coolant exposure tests in the MIT research reactor.

The outer dense layer of SiC serves as an environmental barrier layer assuring the cladding can achieve at least seven years of exposure to PWR coolant without excessive corrosion. Figure 2 below presents the results of coolant exposure tests after 30 months in PWR coolant temperature and chemistry conditions.

Round 6 after 30 months in MIT Reactor

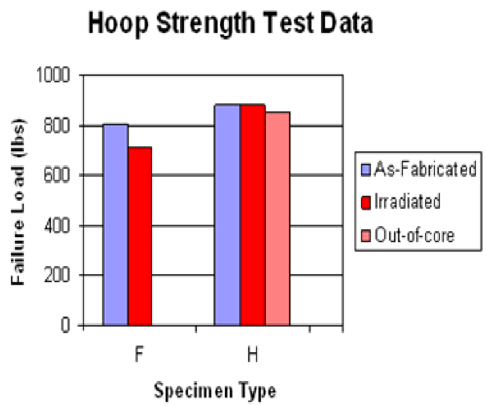

Figure 2 – MIT Reactor PWR Coolant (300 °C) Exposure Test Results

Out-of-pile accident tests have demonstrated the superior behavior of TRIPLEX™ clad during simulated LOCA events. For example, a quench test of SiC TRIPLEX™ clad by emersion of a 1000 °C specimen into a room temperature water pool showed virtually no damage or loss of structure. Other tests in a 1200 to 1400 °C LOCA steam environment for over 8 hours demonstrated the dramatic reduction in exothermic reaction and hydrogen release cited above. A test of fuel and clad interaction at typical PWR operating temperatures and flux is ongoing in the Oak Ridge High Flux Isotope Reactor.

- **Summary at 1200 C for 6 hours**

Zircaloy-2 values are predictions based on well established data;
SiC values are based on measurements at CTP Lynchburg

Hydrogen released– ml /cm2 -	SiC	0.25
	Zircaloy 2	160.7
Recision in microns	SiC	0.72
	Zircaloy 2	530

- **Summary at 1400 C for 8 hours**

Hydrogen released – ml/cm2	SiC	1.5
	Zircaloy 2	170
Recision in microns	SiC	4.2
	Zircaloy 2	890

Figure 3 – SiC Accident Simulation (Steam Exposure) Tests at CTP Lynchburg

Further research is ongoing to (1) develop and demonstrate a reliable high integrity end cap joint, (2) demonstrate a new manufacturing approach for fabricating 14 foot long clad tubes, (3) further improve the fracture resistance of the clad tube, and (4) minimize the potential adverse effect on steady state and transient UO_2 fuel temperature resulting from the absence of creep of the cladding on to the pellet, and resulting pellet clad gap, during normal operation. A solution to each of these issues is near resolution. With some further investment by the international fuel supply industry, and National governments, this technology could be ready for in reactor testing within five to ten years.

In a parallel program, sponsored by EPRI (US Electric Power Research Institute), a mechanical shock test of a BWR channel box model (4" x 4" x 24") fabricated from layers of SiC-SiC composite material was performed in March 2012. The test demonstrated acceptable shock resistance for use in commercial BWRs. Further development of the basic material of construction including fiber architecture as needed to assure adequate dimensional stability during irradiation, and the absence of delamination, is now being pursued. Once this work is completed, introduction of test units into test reactors and then commercial BWR reactors, will be considered.

3. Conclusions

Although the zircaloy cladding used in commercial water reactors for the last fifty years has proved very reliable during normal operation, it has also been the major contributor to the severity of core overheating accidents such as occurred at Three Mile Island and Fukushima. If the industry and the governments who are dependent on nuclear power, are serious about reducing the consequences of such severe accidents, thereby improving the future safety of commercial nuclear reactor operations, then they should immediately invest in the further development, testing and licensing of alternative, non reactive, fuel cladding and channel boxes similar to the unique SiC TRIPLEX™ fuel cladding, and dimensionally stable SiC-SiC channel box, described in this paper.