Severe Accident Modeling of a PWR Core with Different Cladding Materials

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Abstract - The MAAP v.4 software has been used to model two severe accident scenarios in nuclear power reactors with three different materials as fuel cladding. The TMI-2 severe accident was modeled with Zircaloy-2 and SiC as clad material and a SBO accident in a Zion-like, 4-loop, Westinghouse PWR was modeled with Zircaloy-2, SiC, and 304 stainless steel as clad material. TMI-2 modeling results indicate that lower peak core temperatures, less H₂ (g) produced, and a smaller mass of molten material would result if SiC was substituted for Zircaloy-2 as cladding. SBO modeling results indicate that the calculated time to RCS rupture would increase by approximately 20 minutes if SiC was substituted for Zircaloy-2. Additionally, when an extended SBO accident (RCS creep rupture failure disabled) was modeled, significantly lower peak core temperatures, less H₂ (g) produced, and a smaller mass of molten material would be generated by substituting SiC for Zircaloy-2 or stainless steel cladding. Because the rate of SiC oxidation reaction with elevated temperature H₂O (g) was set to 0 for this work, these results should be considered preliminary. However, the benefits of SiC as a more accident tolerant clad material have been shown and additional investigation of SiC as an LWR core material are warranted, specifically investigations of the oxidation kinetics of SiC in H₂O (g) over the range of temperatures and pressures relevant to severe accidents in LWR’s.

I. INTRODUCTION AND BACKGROUND

The current operating fleet of Generation III commercial light water nuclear reactors (LWR) and the new, under-construction Generation III+ LWR’s are all required to survive specific design basis accidents by federal regulation. Design basis accidents, such as a loss of coolant accident (LOCA), are used in the design and subsequent fabrication of both the reactor plant and in-core components. As example, during a postulated LOCA the fuel assembly is required to maintain a coolable geometry and the fuel rods in the assembly are required to retain all fissile material and fission products. For current Generation III and III+ reactors, this generally requires that the reaction between fuel rod clad zirconium (Zr) alloys and high pressure/high temperature steam (H₂O (g)) occurs in a well understood, predictable manner and the fission product barrier is not compromised and core rearrangement does not occur.

The recent accident at the Fukushima Daiichi nuclear power station has renewed attention on the resistance of current and future Generation III and III+ LWR’s to design basis and beyond design basis accidents. In the case of Fukushima Daiichi, a station blackout accident (SBO) occurred in which all on and off site power was lost for an extended period of time. This extended loss of power resulted in reactor core uncover scenarios significantly beyond a design basis LOCA accident and is more correctly termed a severe accident. Due to loss of all power for an extended period of time, almost all cooling water from the core was lost, core fuel assemblies were uncovered and overheated by decay heat which caused the fuel rod clad Zr alloy to react with the remaining H₂O (g). This reaction resulted in loss of fission product barrier, eventual core melt down, and core relocation. (1)

In light of the Fukushima Daiichi severe accident, it is prudent to consider what modifications to reactor core materials could be approached to minimize or altogether avoid the reactions that occurred at Fukushima Daiichi.
The progression of such a severe accident involves complex interaction of several phenomena and the resulting behavior may be somewhat subtle as is the case of hydrogen (H\textsubscript{2} (g)) formation from the Zr alloy - H\textsubscript{2}O (g) reaction. One approach to attempt to minimize the in-core Zr - H\textsubscript{2}O (g) reaction would be to replace the currently used Zr alloy clad material with one that generates significantly less H\textsubscript{2} (g) and heat when heated to high temperatures and exposed to H\textsubscript{2}O (g), and has other properties beneficial in a severe accident. Silicon carbide (SiC) ceramic matrix composite materials have been under consideration for fuel cladding application for some time.

(2) SiC is known to have significant high temperature properties, including a decomposition temperature of 2545°C. (3) Additionally, SiC is not known to react as exothermically and as rapidly as Zr in high temperature H\textsubscript{2}O (g). SiC does react with H\textsubscript{2}O (g) at elevated temperatures, however the reaction under the conditions tested is primarily oxidation generating a silica (SiO\textsubscript{2}) scale and volatilization of this SiO\textsubscript{2} scale. (4,5) Therefore, while not inert under severe accident conditions, SiC should offer significant advantage relative to presently used Zr alloys and should be investigated further.

As a first approach to developing more accident resistant nuclear fuel, this paper presents a study modeling the response of an LWR core in accident scenarios using 3 different materials as cladding; Zircaloy-2, SiC, and 304 stainless steel. The Modular Accident Analysis Program (MAAP v.4) software is used to model the response of these 3 materials in 2 different severe accidents. First, the Three Mile Island reactor 2 (TMI-2) severe accident is modeled using Zircaloy-2 and SiC as clad materials. TMI-2 is a relatively very well understood severe accident and the accident response of core materials other than Zircaloy-2 can be accurately compared. (6) Second, a Zion-like, 4-loop, Westinghouse pressurized water reactor (PWR) is modeled in a SBO accident with these 3 clad materials. By modeling a SBO accident, it is intended that the results provide a first approach at predicting the material behavior in a severe accident similar to what occurred at Fukushima Daiichi.

II. MODELING METHOD AND TECHNICAL APPROACH

The MAAP software is an Electric Power Research Institute (EPRI) owned severe accident analysis code that has been developed to examine the response of Boiling Water Reactors (BWR), Canada Deuterium Uranium reactors (CANDU), and PWRs for conditions that could lead to core damage. MAAP is widely used by fuel vendors, nuclear power generating utilities, and other organizations in Europe, Asia, Canada, and the United States for all of the above reactor designs and the various types of containments that are used for each reactor type.

For each of the reactor types listed above, MAAP models the reactor core, the Reactor Coolant System (RCS also called the Heat Transport System for CANDUs), the containment and any adjacent plant buildings that could be influential for a given accident sequence, such as the reactor building for BWRs or the auxiliary building for PWRs. Of particular interest for this comparison of cladding materials is the response of the core after the fission reaction has been shut down but an accident condition exits that causes depletion of the water inventory cooling the core. Two accident conditions that could cause this depletion are a LOCA and a Loss of Heat Sink (LOHS) as could occur during a SBO. With water depletion, the water level would eventually drop below the top of the core and the temperatures in the uncovered part of the core would begin to increase. As the Zr cladding temperature increases, the rate of oxidation in the steam environment increases, with H\textsubscript{2} (g) being released by the reaction.

To model the axial and radial power distribution and the resulting transient temperature distribution as the water level decreases, MAAP models the core as axially nodalized radial regions with five radial regions being used in the PWR analyses for this work. As the water level is depleted, the top of the vertical core is uncovered first and heats up according to the decay power in that node combined with any heating or cooling by the steam that is convected from below. The steaming rate in each radial region is determined by the decay heat that is generated under the boiled-up water level. In an open lattice PWR core, the coolant can flow between the fuel assemblies and hence there is only one water level. As this water level decreases, the fuel and cladding temperatures increase and this combined with any H\textsubscript{2} (g) that is generated by the local cladding oxidation causes the gases in the core region to be less dense than those in the upper plenum and this initiates a natural circulation flow that tends to rise in the center and fall downward in the outer most core region. This natural circulation is superimposed on the upward steam flow rate that is caused by the decay heat in the submerged part of the core. These flows are all represented in the MAAP code and have been compared to those one-seventh scale integral experiments that have been sponsored by EPRI, the Nuclear Regulatory Commission (NRC), and performed in the Westinghouse Laboratories (6,7,8).

For this evaluation of different core materials, comparative evaluations of the same reactor configuration were performed with different cladding materials. For these comparative analyses, MAAP evaluations were performed for a SBO accident sequence in a Zion-like 4-loop, Westinghouse PWR. To represent the behavior of the SiC cladding, the properties of the average density, specific heat, thermal conductivity and melting point were added to the code. As a preliminary comparative study of the
behavior of the two materials, the SiC evaluation is assumed to experience no chemical reaction with the superheated steam. This provides the maximum possible benefit for the SiC material in this comparative analysis.

A second means of comparing the behaviors of the different cladding materials is to examine the response of the TMI-2 accident, again assuming that cladding materials are Zircaloy in one case and SiC in the other. This specific accident sequence is of interest since most of the core was uncovered for tens of minutes causing the fuel rods to be overheated such that severe oxidation of the Zircaloy cladding occurred. Post accident evaluations concluded that as much as 1000 lbm of $\text{H}_2$ (g) was created during the accident. (9) The reaction heat liberated by the oxidation reaction was the principle reason that the core melted in this relatively short interval. Since the TMI-2 accident experienced reflooding of the reactor core, it is of interest to compare the calculated response of two different cladding materials for this accident sequence as well.

### III. MODELING RESULTS

#### III.A. TMI-2

The TMI-2 severe accident was a partial core melt down accident occurring in 1979. The significance of this accident to nuclear reactor design and safeguards is that it was forensically thoroughly analyzed. (10) With such detailed analysis available, modeling of this accident scenario with different core cladding materials can result in very accurate determination of the response of these different materials under well known accident conditions.

TMI reactor unit 2 was a Babcock & Wilcox design BWR with once through steam generators. The sequence of events during the TMI-2 accident is generally as follows;

1. Core uncovered ~120 minutes into the accident.
2. Peak core temperature attained ~150 minutes into the accident.
3. Core reflooded ~174 minutes into the accident.

(11)

From this general event and time guideline, MAAP v.4 was used to model TMI-2 accident progression with Zircaloy-2 and SiC as fuel cladding materials. Again, for this model scenario SiC is assumed to experience no chemical reaction with the superheated steam. While it is known that SiC will experience some reaction with $\text{H}_2\text{O}$ (g) at elevated temperatures, the heat and $\text{H}_2$ (g) generated from this reaction should be significantly less than that with Zircaloy-2. Therefore, as a preliminary approach, assuming no SiC - $\text{H}_2\text{O}$ (g) reaction is warranted.

Figure 1 presents the temperature in the hottest core node during the TMI-2 accident model. Zircaloy-2 is seen to attain $2870^\circ\text{C}$ peak temperature at 2.9 hours (2 hrs. 54 min.) due to first decay heat and then the heat generating exothermic $\text{Zr} - \text{H}_2\text{O}$ (g) reaction. SiC attains a peak temperature of $1200^\circ\text{C}$ at a similar time due only to decay heat assuming no reaction of SiC with elevated temperature $\text{H}_2\text{O}$ (g). Due to excellent high temperature stability, if SiC would attain a peak core temperature near $1200^\circ\text{C}$ during a TMI-2 accident scenario, then the cladding as fission product barrier would not have been compromised. Note that the SiC temperature is modeled to decrease at time greater than 2.9 hours because the core is reflooded. It should also be noted that the core node with the maximum temperature can change as the accident progresses.

Figure 2 presents the mass of $\text{H}_2$ (g) generated during the MAAP v.4 model of the TMI-2 accident scenario. Due to the oxidation of Zircaloy-2 in high temperature $\text{H}_2\text{O}$ (g), approximately 1000 lb. mass of $\text{H}_2$ (g) is produced 3 hours into the accident. Comparatively, SiC generates no $\text{H}_2$ (g) again due to the assumption of no reaction with elevated temperature $\text{H}_2\text{O}$ (g).

Figure 3 shows the RCS pressure, which except for the comparatively small static head differences is essentially uniform throughout the RCS. Again, due to the reaction of Zircaloy-2 with high temperature $\text{H}_2\text{O}$ (g) and subsequent generation of a large mass of $\text{H}_2$ (g), a maximum pressure of 2300 lb/in$^2$ is produced approximately 3 hours into the accident. SiC cladding shows effectively no increase in RCS pressure due to no reaction with high temperature $\text{H}_2\text{O}$ (g).

Figure 4 presents the modeled mass of molten material generated in the TMI-2 accident scenario. Due to a peak core temperature of $2870^\circ\text{C}$, figure 4 shows that up to 68,000 lb. of molten material would be produced in a Zircaloy-2 clad core. This mass of molten material would consist of both melted fuel rods and other in-core components. For a SiC clad core, figure 4 shows that only 3000 lb. of molten material would be produced, with all of this mass consisting of in-core components other than fuel rods. Figure 1 shows that SiC would attain $1200^\circ\text{C}$ peak temperature in a modeled TMI-2 accident. This is why no fuel rods are part of the molten mass produced during the accident.
Fig. 1. Temperature in the hottest core node for Zircaloy cladding (blue solid line) and SiC cladding (red dashed line) during a modeled TMI-2 accident scenario. TCRHU is hottest core node temperature in °F.

Fig. 2. Mass of hydrogen produced for Zircaloy cladding (blue solid line) and SiC cladding (red dashed line) during a modeled TMI-2 accident scenario. MH2CR is total mass of H₂ (g) generated in the core in lb.

Fig. 3. Reactor cooling system pressure generated for Zircaloy cladding (blue solid line) and SiC cladding (red dashed line) during a modeled TMI-2 accident scenario. PPS is the reactor cooling system pressure in lb/in².

Fig. 4. Mass of molten material generated for Zircaloy cladding (blue solid line) and SiC cladding (red dashed line) during a modeled TMI-2 accident scenario. MLTCR is molten mass in the core in lb.

III.B. Station Blackout

The SBO sequence examined is one in which there is a total loss of all on-site and off-site AC power, but DC batteries are available to power the instrumentation. In addition, it is assumed that there is no Auxiliary Feedwater (AFW) available for injection to the steam generators. Hence, there is no heat removal from the RCS. Once the steam generators dryout (about 100 minutes), the RCS begins to pressurize until the safety valve opens and this releases the high temperature RCS water to the pressurizer drain tank (located in the containment building). With this
discharge from the pressurizer, the drain tank connected to the pressurizer is eventually pressurized sufficiently to break the rupture disk. After this occurs, the RCS coolant is discharged into the bottom of the containment building as a steam-water mixture.

As the loss of RCS cooling water continues, the top of the fuel is eventually uncovered. When the core is approximately one-third uncovered, the temperature at the top of the core approaches a value where the steam oxidation rate of Zr cladding produces a chemical energy release that is comparable to the average decay heat generated in the fuel. Furthermore, at this condition, natural circulation begins to occur between the core and the upper plenum due to the lower density, hotter gases in the core region compared to those in the upper plenum. Additionally, natural circulation flows would be established between the upper plenum and the steam generator inlet plenum along with circulatory flows through some of the outbound inverted U-tubes to the steam generator outlet plenum with return flow through other inverted U-tubes (see Figure 5). This flow through the steam generators also includes a plume of hot gases that would ascend to the bottom of the steam generator tube sheet and this plume would determine the hottest temperature that would be subjected to the steam generator tubes as a result of these natural circulation processes (see Figure 6). These natural circulation flows are included in the MAAP models.

These natural circulation flows increase the steam mass that is available to continue the cladding oxidation process for the Zircaloy clad design. For substitution of SiC cladding in the core, these natural circulation processes also act to distribute the heat load from the core to other parts of the RCS including the upper plenum structures, the hot legs, and the steam generators. Nevertheless, the rate of energy distribution is considerably greater with Zr cladding because of the runaway chemical reaction associated with the core melt progression.

It is also noted that the lowest melting temperature materials in the core are the silver-indium-cadmium control rod material, the stainless steel and nickel alloy tubing used for the control rod fingers, and the in-core instrument tubes. It is further noted that the in-core instrument tubes have a central hole that is used for the core flux monitors to have access to different parts of the core. This central hole is at containment pressure and with the melting of the surrounding instrument tube structures, this hole functions as a release path for both H₂ (g) and the gaseous/aerosolized fission products. The evidence from the TMI-2 accident indicates that this central opening was the path for radioactive fission product releases to the containment. (12)

Table 1 lists the behaviors of the two different types of cladding material with respect to the timing of when the core is uncovered and also for the first breach of the RCS boundary. As expected, the time when the core water level decreases below the top of active fuel (core is uncovered) is unaffected by the cladding material. The next SBO accident time of interest is when the model calculates a challenge to the RCS integrity, which is a rupture of one of
the hot legs due to the natural circulation between the overheated core and the upper plenum, the upper plenum and the cooler steam generator inlet plenums as discussed above.

Table I
SBO Results for a Zion-Like Design

<table>
<thead>
<tr>
<th></th>
<th>Zircaloy-2 Cladding</th>
<th>Silicon Carbide Cladding</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core is Uncovered</td>
<td>8655</td>
<td>8656</td>
</tr>
<tr>
<td>Time to Hot Leg Creep (sec)</td>
<td>12874</td>
<td>13965</td>
</tr>
</tbody>
</table>

Figure 7 illustrates the hottest core node temperature developed for the two different types of cladding in a SBO accident as modeled using MAAP. Again, this is not necessarily the same node throughout the temperature transient. As illustrated, the Zircaloy-2 cladding begins rapid oxidation at approximately 12,500 seconds as indicated by the rapid rise in the hottest core node temperature. Conversely, with the lack of an exothermic oxidation reaction, there is no rapid temperature increase with the silicon carbide cladding material. It is also noted that the difference between the Zircaloy-2 and SiC behavior begins at approximately 10,000 seconds and this difference is due to the evolving oxidation history of the Zircaloy cladding.

It is also noted that at temperatures of approximately 1300 to 1400 K, there is no significant strength left in the stainless steel core components, especially the in-core instrument tubes. As these are either failed due to the imposed external pressure difference between the RCS and containment pressures or due to melting, these instrument tubes would then become a path to release fission products to the containment environment. Furthermore, these would then be part of the early-on relocation of core material from the top of the core downward to the cooler regions in the middle and bottom of the core.

The time of hot leg creep rupture is also noted in Figure 7. As discussed above, this results from the onset of natural circulation flows between the core and the steam generators. Nonetheless, the creep failure of the hot leg is a time-at-temperature phenomena that is caused by the high temperature flows from the vessel to the steam generators flowing along the top of the hot leg. The colder return flows would move along the bottom half of the pipe. While the temperatures are lower with the SiC cladding, the hot leg failure is calculated to occur approximately 20 minutes later. Therefore, for an unmitigated SBO accident sequence, the same challenge will occur for the RCS integrity, but on a somewhat different timescale.

![Fig. 7. Comparison of the hottest core node temperature for Zircaloy cladding (blue solid line) and silicon carbide cladding (red dashed line) for a Zion-like, 4-loop, Westinghouse PWR SBO accident sequence as modeled using MAAP. TCRHOT is hottest core node temperature in K.](image)

Hot leg creep rupture would rapidly depressurize the RCS and this dramatically reduces the steam density and the oxidation rate. Consequently, this phenomenon somewhat masks the behavior that may occur for other possible accident sequences which could progress at an elevated pressure, but not as high as that calculated for the SBO. To provide a better surrogate for examining the influence of cladding materials, the SBO is used with the RCS depressurization following creep rupture disabled. This was done in MAAP by setting the rupture area to zero. It is understood that this is non-mechanistic, but it is helpful to see the true potential for cladding oxidation.

As part of these preliminary investigations, questions have been asked with regard to:

1. Would stainless steel provide for a substantially different accident sequence behavior than Zircaloy-2, and
2. How would the stainless steel cladding behavior compared to that of SiC?

MAAP v.4 includes stainless steel cladding in a steam environment using the reaction kinetics reported by Wilson (13,14) and those of White (15). As part of these observations, it was noted by Wilson that the stainless steel oxidation is complicated by the fact that the steel samples exhibited swelling and foaming as the metal approached temperatures of 1400°C and this led to substantial uncertainties in the reacting surface area. Also, in his experimental studies, White concluded that the initial oxidation rate demonstrated a linear behavior which was followed by a parabolic behavior at higher temperatures.
To illustrate the underlying behaviors associated with the various cladding materials, the three types of cladding were analyzed for a SBO accident sequence without any recovery actions. Figure 8 illustrates the temperature of the hottest core node for the three different cladding materials. As expected, all three cladding materials began to experience heat up at the same rate as a result of the core being uncovered at approximately 8,700 seconds. The uniform heating rate continues for approximately 1500 seconds at which time the Zr cladding calculation exhibits an increasingly faster heating rate as a result of the accelerated oxidation, which results in a runaway chemical reaction at approximately 12,500 seconds. In comparison, both the stainless steel and SiC materials exhibit nearly the same heating rate until approximately 12,000 seconds when the stainless steel exhibits a faster heating rate as a result of the cladding reaction with steam. As expected, this heating rate for the stainless steel begins to increase after approximately 12,000 seconds. SiC cladding material exhibits a decreasing heating rate after approximately 12,000 seconds because it is assumed to have no exothermic chemical reaction.

Figure 9 shows the calculated hydrogen mass generated as core degradation progresses for the three different cladding materials. As discussed for Figure 8 the Zircaloy-2 cladding generates approximately 470 kg of H2 (g) with the major fraction being generated by the runaway chemical reaction beginning at approximately 12,500 seconds. Once the Zircaloy material melts, it also begins to liquefy the fuel and eventually the molten core materials begin to relocate to the lower parts of the reactor core and subsequently into the reactor pressure vessel lower plenum. Core relocation is why the H2 (g) generation effectively stops for the Zircaloy-2 cladding evaluation. With slower reaction kinetics, the stainless steel material does not form H2 (g) as rapidly early in the accident sequence. This is consistent with the fact that the stainless steel clad fuel had more margin related to the peak clad temperature for design basis evaluations. Nevertheless, for this accident sequence with long term core uncover and substantial overheating, the stainless steel cladding eventually produces more H2 (g) than the Zircaloy cladding. One of the differences between the Zircaloy and stainless steel oxidation behavior is that the Zr steam reaction produces two moles of hydrogen for every mole of Zr reacted, whereas in a hydrogen-rich atmosphere the iron and chromium constituents of stainless steel produce four moles of hydrogen for every mole of iron reacted. Thus, when comparing different candidate cladding materials, one needs to take into consideration both the kinetics of the reaction and oxidation reaction products.

Lastly figure 10 presents the molten mass produced in the reactor core during an extended SBO scenario. As expected, the zirconium and stainless steel cladding materials have nearly the same production rate of molten materials within the core. The SiC cladding material has only a limited extent of molten material within the timeframe examined for this accident sequence. This illustration of the development of large amounts of molten materials in the core directly reflects the released energy as a result of the chemical reaction between the different cladding materials and steam.
Fig. 10. MAAP calculated molten mass within the core for the different cladding materials for a station blackout in a PWR reactor with the hot leg creep rupture model disabled.

IV. INTERPRETATION AND SIGNIFICANCE OF MODELING RESULTS

The modeling results presented show significant relative benefit of applying specific materials with enhanced tolerance to severe accidents in PWR cores. Results such as in figures 1, 4, 8, 9, and 10 should be relatively interpreted that there are materials available that could better withstand PWR severe accident scenarios than presently used Zr base alloys. Dilute Zr base alloys have been successfully used as PWR fuel cladding materials for decades. However, as has been known and regulated to also for decades, this alloy system is intolerant of core uncover accident scenarios; i.e.; reaction with high temperature H₂O (g). (16) From this study it should be interpreted that materials, such as SiC in this instance, could provide modified thermo-physical and chemical properties that would prove more tolerant in PWR core severe accident scenarios relative to presently used Zr alloys.

Critical factors in identifying and evaluating materials exhibiting enhanced accident tolerance should include relative normal operating condition behavior in a PWR core and the determination and application of specific materials properties. For this work, the oxidation reaction rate of SiC with elevated temperature H₂O (g) was set to 0. This was done for preliminary evaluation and ease of application of the MAAP code. It is understood that, when exposed to elevated temperature H₂O (g), SiC will oxidize. (5) However, reaction rate kinetics in pure H₂O (g) and covering the desired temperature range (~500 to 1800°C) are not fully available. Additionally, the rate of oxidation of SiC in elevated temperature H₂O (g) will be significantly lower than that of Zr alloys. Therefore, the modeling results of this study offer relative significance and should be used as guidance for future investigations, such as determination of the oxidation kinetics of SiC in H₂O (g) over a range of temperatures and pressures relevant to severe accidents in PWR’s.

V. CONCLUSIONS

The MAAP v.4 code has been used to model the TMI-2 severe accident and a SBO severe accident as applied to a Zion-like, 4-loop, Westinghouse PWR using Zircaloy-2, SiC, and type 304 stainless steel as cladding materials. Required thermo-physical and chemical properties of these 3 materials are already in MAAP or were input with modification. For SiC, this property modification involves setting the rate of oxidation reaction with elevated temperature H₂O (g) to 0. Due to this property modification, this work and these results should be considered preliminary.

Modeling results show that by substituting SiC for Zircaloy-2 or stainless steel, enhanced accident tolerance could be obtained in the event of an LWR severe accident. In the case of the TMI-2 severe accident, modeling results show that substituting SiC for Zircaloy-2 cladding would result in lower peak core temperatures, less H₂ (g) produced, and a smaller mass of molten material generated. For a SBO severe accident, modeling results show that substituting SiC for Zircaloy-2 cladding would increase the calculated time to RCS rupture by approximately 20 minutes. Finally, for an extended SBO severe accident with hot leg creep ruled out, modeling results show that substituting SiC for Zircaloy-2 or stainless steel cladding would result in significantly lower peak core temperatures, less H₂ (g) produced, and a smaller mass of molten material generated. Based on these preliminary results, it is offered that additional investigation of SiC as an LWR core material be undertaken with specific work on the oxidation kinetics of SiC in H₂O (g) over the range of temperatures and pressures relevant to severe accidents in PWR’s.

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NOMENCLATURE

LWR – Light Water nuclear Reactor
LOCA – Loss of Coolant Accident
SBO – Station Blackout accident
MAAP – Modular Accident Analysis Program
TMI-2 – Three Mile Island reactor 2
PWR – Pressurized Water Reactor
EPRI – Electric Power Research Institute
BWR – Boiling Water Reactor
CANDU – Canada Deuterium Uranium reactor
RCS – Reactor Cooling System
LOHS – Loss of Heat Sink
NRC – Nuclear Regulatory Commission
AFW – Auxiliary Feed Water

REFERENCES