Transactions of the Korean Nuclear Society Autumn Meeting
Jeju, Korea, October 21-22, 2010

Preliminary Acceptance Criteria for Safety Analysis of KALIMER-600 SFR

Young-Min KWON [*], Kwi-Lim LEE, Kwi-Seok Ha, Won-Pyo Chang, and Hae-Yong JEONG
Korea Atomic Energy Research Institute, 1045 Daedeok St.,Yuseong, Daejeon 305-353 KOREA
[*]*Corresponding author: ymkwon@kaeri.re.kr*

1. Introduction

The KALIMER-600 event categorization in the function of occurrence frequency has been made by traditional engineering judgment with information from some reference plants such as CRBR, PRISM [1] and EFR. The dividing line between DBE and BDBE is the frequency of 10^{-7} per plant-year. Each event belongs to one of five categories based upon its nominal frequency per reactor-year (f) as a criterion.

(1) Moderate frequency Event (MF): $f \geq 10^{-1}$
(2) Infrequent Event (IE): $10^{-1} > f \geq 10^{-2}$
(3) Unlikely Event (UE): $10^{-2} > f \geq 10^{-4}$
(4) Extremely Unlikely Event (XU): $10^{-4} > f \geq 10^{-7}$
(5) Beyond DBE (BDBE): $> f \geq 10^{-4}$

2. Acceptance Criteria for Safety Analysis

The ultimate goal of the safety analysis is to show that radiological releases to the plant personnel in the site and to the environment are limited. For each DBE category the radiological limits need to be specified according to the regulation guide. Although the safety criteria for the event conditions within DBE are the radiological limits, alternative physical limits which ensure that the radiological limits are not exceeded have been used for the KALIMER-600 safety analyses because they could be more easily applied to the design process [2]. The key phenomena-based criteria for MF, IE, UE and XU are presented in Table 1.

The acceptance criteria presented in Table 1 are based on the premise that if appropriate fuel design and coolable geometry limits are not exceeded and if radiological releases are limited so that the dose guidelines are not exceeded, then the public health and safety are adequately protected. Conservative and quantifiable criteria are set based on current knowledge of irradiated metal fuel and HT9M cladding pin behavior to ensure that the safety requirements are met. The currently accepted temperatures addressing the safety limits in Table 1 are summarized in Table 2, where the temperature limits for the core outlet coolant, cladding, coolant within subassembly, and fuel are presented.

The key phenomena-based criteria for ME, IE and UE are:

- No local fuel melting
- No fuel-cladding liquid phase formation
- Cladding strain and creep rupture damage fraction limited to preclude pin failure
- No sodium boiling
- Reactor structural integrity assured

A similar set of the safety criteria for XU and BDBE are listed below. Overall, the criteria set are more conservative in aspect related to public safety; however, the transients are evaluated on a nominal basis. The criteria are consistent with the NRC's requirements proposed for the PRISM design.

- Local fuel melting without + reactivity addition
- Limited cladding failures with no propagation
- No sodium boiling
- Maintenance of primary boundary integrity

3. Physical Basis for Acceptance Criteria

3.1 Structural Integrity Criteria and Limits

Reactor structural integrity is essential to assure core cooling and radiological containment. Quantification of structural integrity uses the ASME Pressure Vessel Code methodology and criteria. The ASME Code specifies a single creep-fatigue damage limit for the entire component life. Translating this limit into separate temperature limits for different operation levels requires apportionment of the damage limit into allowable damages for the individual operation levels.

The loading and time evaluations for projected event scenario are based on the simplified structural analysis where a considerable conservatism was included in the damage apportionment and the component environmental conditions.

3.2 Cladding Damage Criteria and Limits

For normal operation (MF) cladding loading, the cumulative damage function (CDF) value is limited to less than 0.001 to empirically assure a low probability of cladding failure. The CDF exceeds the limitation of 0.001 when the cladding temperature becomes higher

Table. 1 Physical Background for Acceptance

Event Category	Avg. Core Exit Temperature	Peak Cladding Temperature		Peak Fuel Temperature	Peak Coolant Temperature	
					Pump on	Pump off
Moderate Frequency Events	Max. Stress, Creep Rupture Time, Service Level A	Limit for insuring burn-up goal	cladding midwall	fuel solidus temperature	Sodium Boiling in the core	Sodium Boiling in the core
Infrequent Events	Max. Stress, Creep Rupture Time, Service Level B	Prevention of rapid cladding integrity degradation				
Unlikely Events	Max. Stress, Creep Rupture Time, Service Level C	Threshold temperature of eutectic melting	Fuel-cladding interface			
Extremely Unlikely Events	Max. Stress, Creep Rupture Time, Service Level D					

Table. 2 Acceptance Criteria for Safety Analysis

Event Category	Avg. Core Exit Temperature		Peak Cladding Temperature		Peak Fuel Temperature (°C)	Peak Coolant Temperature (°C)	
	Temperature (°C)	Allowance time (hours)	Temperature (°C)	Allowance time (hours)		Pump on	Pump off
Moderate Frequency Events	560	≤ 40,000	< 650	≤ 52,000	955	1,055	940
Infrequent Events	600	≤ 1,000	650 - 670	≤ 240			
Unlikely Events	650	≤ 30	< 700	0			
Extremely Unlikely Events	700 760	≤ 5 ≤ 1					
ATWS	700 760	≤ 5 ≤ 1	< 700 < 790	≤ 1.2 ≤ 0.3	1,070	1,055	940

than 650°C.

Rapid cladding integrity deterioration begins to occur with drastic decrease of creep resistance due to the stress exponent increases as operating temperature exceeds 670°C at 80MPa. The allowable holding time is assumed 240 hours. During UE and XU transients, the cladding temperatures must remain below the eutectic temperature (700°C) of the HT9 alloy formed by fuel and cladding inter-diffusion.

Cladding rupture is the principal fuel failure mechanism that release fission products and fuel into the coolant. Considering cladding wastage and the transient temperature history, cladding creep rupture is the appropriate mechanistic criterion applied to quantify the cladding integrity criterion. A limited cladding failure without failure propagation is allowed for XU and BDBE events. The fuel-cladding attack is limited to less than 10% of the cladding wall thickness in order to limit the strength degradation and the amount of fuel liquefied.

The inner diameter wastage from molten fuel-cladding alloy attack at different fuel-cladding interface temperature conditions was calculated considering the actual time history in Fig. 1. An early fuel rod failure due to creep rupture occurs at about 1080 seconds (0.3 hours) when the cladding maintains a constant temperature of 790°C. Short term excursions above the alloy solidus do not result in cladding failure and are permitted if the cladding thinning is limited to 10%.

3.3 Fuel Melting Criteria and Limits

Based on the recent metal fuel tests, centerline fuel melting, even extensive melting exceeds 80% of a given cross-section, is not a problem and does not result in pin failure. Figure 2 illustrates the U-Pu-Zr fuel melting temperature as a function of Pu contents.

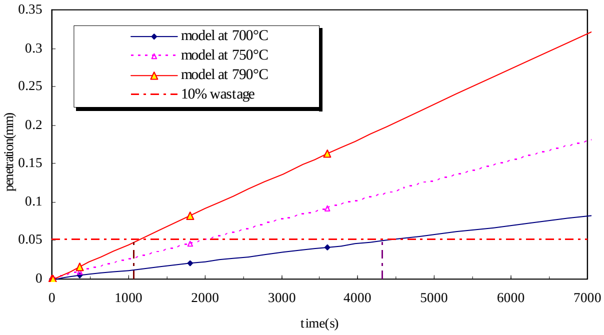

Fig.1 Penetration Rate of HT9 Cladding

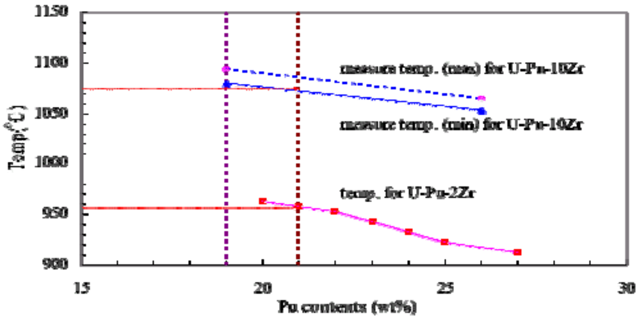

Fig.2 Fuel Melting Temperature Limit

During normal operation and DBE transients, the peak fuel temperature, including uncertainties, must remain below the fuel melting (solidus) temperature considering local constituent migration. Zirconium migration creates a depleted region within fuel and blanket in which the fuel melting temperature is degraded. The currently assumed temperature limit for this criterion is 955°C for the DBE category.

More permissive but still conservative criterion is adequate for BDBE or ATWS events. Even though centerline molten fuel under overpower transients does not contribute to cladding failure, limitation of the amount and time duration of molten fuel is given by that less than 50% of pin cross-sectional area is melt during less than 2 minutes. Because the principal issue with centerline fuel melting is the possible separation of fission gas bubbles from the molten fuel which would result in fuel compaction and possible positive reactivity addition. The equivalent fuel temperature greater than 1070°C for less than 2 minutes eliminates the potential of fuel motion reactivity effects.

3.4 Coolant Boiling Criteria and Limit

Protection of the reactor vessel or internal structures is provided by limiting the potential for the ATWS event to progress into energetic events. Because U-Pu fuel has a positive coolant voiding reactivity coefficient over much of the active core length, significant boiling must be avoided. Local boiling in the core results in an increased cladding failure rate as cladding and fuel-cladding interface temperature increase.

The peak coolant temperature within a subassembly must be less than the local saturation temperature. When the primary pumps do not operate to pressurize the core, the pressure at the top of the active core is 0.1651MPa and the corresponding saturation temperature is 940°C. During full core flow events with the primary pumps operating, the pressure is 0.3902MPa and the saturation temperature is 1055°C.

4. Concluding Remarks

The alternative physical limits which ensure that the radiological safety limits are not exceeded have been used for the KALIMER-600 safety analyses because they could be more easily applied to the design process. However, it should be emphasized that the numeric limits for the criteria are subject to change as KALIMER-600 design progresses; however, the criteria stated in terms of the physical phenomena are not expected to change.

References

[1] US. NRC, NUREG-1368, February 1994.

[2] Y. M. Kwon et al., “Inherent Passive Safety Characteristics of a Sodium-Cooled, Metal-Fueled, Pool-Type Fast Reactor,” KAERI/TR-3727/2009.