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SFR INHERENT SAFETY FEATURES AND CRITERIA ANALYSIS

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This study presents the inherent safety performance of BN-type reactor 2800 MW of thermal power with MOX core during ATWS initiated by various accident initiating events with simultaneous failure of all shutdown systems in all cases under investigation.
The most severe cases leading to the pin cladding rupture and possible sodium boiling were found out.
The impact of various safety features on SFR inherent safety performance during ATWS was also analyzed. The decrease in hydraulic resistance of primary loop, increase in primary pump coastdown, the implementing of thermo-mechanical, leakage based and other self-actuated safety systems considered as additional natural feedbacks were considered. Performing analysis resulted in a set of recommendations to the characteristics of the features referred above for the purpose of enhancing the inherent safety performance of SFR under investigation.
In order to exclude the safety barrier rupture during ATWS the set of criteria defining the ATWS processes dynamics and requirements to them were recommended based on achieved results. These criteria are include the natural circulation onset level (must exceed 0.07 rel. units in most severe case), the coolant flow rate drop under the natural circulation onset level (must be missing 0.015 rel. units in most severe case) and time from coolant flow rate drop under the natural circulation onset level till natural circulation onsets (must be missing 101 s). The recommendations for way to implement the self-actuated safety systems are also elaborated.
The analysis of admitted assumptions and obtained results revealed that to develop the refined requirements for the proposed criteria it is necessary to couple the SFR performance analysis for ATWS with uncertainty analysis. It is also necessary to take into account heat removal through passive heat removal systems even in a failure mode by heat-conductivity through the HX walls and to refine the acceptable temperatures of critical components of reactor (fuel, cladding, coolant and reactor tank) with respect to reactor inherent safety. The suitability of chosen acceptable temperatures values of critical components of reactor is discussed.
The results of the inherent safety analysis presented in this study are obtained by using the one-dimensional DYANA code for inherent safety analysis of fast liquid metal cooled reactors. Estimated sodium temperature and mass flow obtained from LOHS+LOF analysis via DYANA code were in reasonable agreement with those obtained from PHENIX benchmark end-of-life test.

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