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REVIEWING THE PIUS REACTOR DESIGN WITH FAILURE MODES EFFECTS AND CRITICALITY ANALYSIS*

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Overview

Advanced reactors proposed for electric power generation in the next century are "evolutionary", meaning that they are improvements on the present generation of light water reactors (LWRs) or "revolutionary" indicating the use of principles for which there is little experience and regulatory precedence. Outstanding in the revolutionary category is the PIUS (Process Inherent Ultimate Safety) conceived by K. Hannerz[1]. PIUS-600 is a 600 MWe pressurized water reactor with no control rods, no active ECCS, a prestressed concrete reactor vessel (PCRV) and designed to operate in a pressure suppression containment with the primary circulation by "wet" variable-speed pumps. Brookhaven National Laboratory, assisting the USNRC in identifying safety-significant aspects of the design, has assembled a multi-discipline team consisting of experts in thermal-hydraulics, PRA, seismic, structural, material and ALARA for this examination. The systems analysis is being done two-fold: failure modes effects and criticality analyses (FMECA)[2] and hazards and operability study (HAZOP)[3]. It is believed that the former "bottom-up" and the latter "top-down" methods complement each other to enhance completeness. This paper describes the FMECA methods that have been applied.

PIUS Operating Principles

The essence of PIUS passive safety is the fact that coolant flow can be directed by the continuity equations i.e. it is not necessary to use active valves to actuate an ECCS. PIUS implements this concept as a primary consisting of a natural circulation path through a borated-water pool and a four-loop pumped flow path through steam generators. When the convective flow of the coolant, resulting from core heating, equals the flow through the four steam generator loops, no flow is possible to the pool. However, any plant disturbance

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that upsets this flow balance results in convective pool flow which, because of the boration, scrams the reactor as well as cools the core. Heat buildup in the pool water is removed by passive convective coolers to the atmosphere.

FMECA Methodology

To discover weaknesses in this robust design, FMECA techniques similar to those being used by BNL in support of the Advanced Neutron Source reactor being designed at ORNL were adopted. This begins with a plant taxonomy which, for reasons of communications and completeness, is the same numerical identification as that used by the vendor (ABB-Atom and Combustion Engineering). These link with system safety descriptions prepared from available documentation. The descriptions are linked with their FMECAs prepared in a specially developed form presented as Table I. This form, prepared using a word processor, identifies the system/subsystem/component being analyzed, for a failure mode/cause resulting in safety impacts on the plant, having a "criticality" of high, medium, low for which a probability, using vendor categories is assigned. The analyst also notes mitigating factors and provides comments and notes to clarify the conditions of the assumed failure.

Completeness and Multiple Failures

Completeness is approached by applying FMECA to all of the systems of the taxonomy deemed to be safety significant. Multiple failures of a deterministic nature are addressed directly in the FMECA by noting the necessity of any support utilities. Multiple random failures are addressed to second order in the FMECA. Higher order random failures are only selectively considered. If this is not sufficient, an interaction matrix will be used.

Results

The results of this work are to identify certain aspects of the plant having safety implication for discussion and resolution. Examples of preliminary discussion items are:

- Redundancy in primary pump control circuitry for scram,
- Control of gas in the gas cap (used to block pool flow during startup),
- Location of the emergency control room,
- Control of non-condensible gases.

Additional discussion item will be presented.

References

1. K. Hannerz, "Towards Intrinsically Safe Light Water Reactors," ORAU/IEA-83-2(M)-Rev. Institute for Energy Analysis, 1983.
2. R. Fullwood and R. Hall, Probabilistic Risk Assessment in the Nuclear Power Industry, Pergamon Press, Oxford, 1988.
3. F.P. Lees, ed., Loss Prevention in the Process Industries, Butterworths, London, 1980.

Table I PIUS 600 Failure Modes and Effects and Criticality Analysis

System: Pressurizer: 328 Reference: PIUS PSID Preparer: R. Fullwood, BNL Date: September 10, 1991

No.	Component/Function	Failure Mode	Failure Cause	Failure Effects (sys/plant)	Criticality	Frequency	Mitigation	Remarks
5	pressure relief valves/ relieve excess pressure	failure to open	corrosion, binding, bad adjustment	failure to releave pressure could lead to dome failure and large LOCA	Low	III	other valves could relieve the pressure	A systematic mis-calibration error could be a common cause leading to over pressure.
6	pressure relief valves/ relieve excess pressure	failure to close	corrosion, binding, poor maintenance	same as a small LOCA	Low	II	none	The conditions and blowdown rate are not expected to result in fuel damage
7	water drain pipe/ drain water to control boron concentration	pipe break	see above previous table	small LOCA	Low	III	none identified	water pipes should be inspectable
8	level sensor/ provide indicators to the steam supply	sensor fails	wiring, electronics, calibration	inability to determine and control water level in pressurizer. Likely forced shutdown.	Low	II	none	continued operation may result in filling pressurizer dome to form a "hard" system. It this is followed by a fluid expansion, the pressure could be damaging.
9	temperature sensor/ monitor steam temperature	sensor fails	see above	reduced knowledge about pressurizer	Low	II	none	

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