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Title of Summary Implementation of an Individual Plant Examination (IPE) at a Nuclear Utility

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Lloyd W. McClure  
ANS Technical Program Chair  
1989 Winter Meeting

**IMPLEMENTATION OF AN INDIVIDUAL PLANT EXAMINATION (IPE) AT A NUCLEAR UTILITY, J. S. Miller and N. G. Cathey (Gulf States Utilities)**

Since the Three Mile Island (TMI) accident which occurred in March 1979, the Nuclear Regulatory Commission (NRC) has implemented numerous regulatory actions. The intentions of these actions are to reduce the likelihood of nuclear power plant accidents that have the potential to impact the health and safety of the general public.

On August 8, 1985, the NRC issued the Commission policy statement on severe accidents in nuclear power plants. In this policy statement the Commission concluded, based on available information, that existing plants pose no undue risk to the public's health and safety. However, the Commission noted that based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), a systematic examination is beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. Therefore, on November 23, 1988, the NRC issued Generic Letter No. 88-20 entitled "Individual Plant Examination For Severe Accident Vulnerabilities". The Individual Plant Examination (IPE) provides a quantification of core damage frequency (CDF)

and fission product release frequency (FPRF). The purpose of quantifying the overall probabilities of core damage and fission product releases is to ensure that the plant specific analyses show the mean core damage frequency (CDF) to be less than 1 in 100,000 per year of reactor operation and the overall frequency of a larger release of radioactive materials to the environment from a reactor accident to be less than 1 in 1,000,000 per year of reactor operation.

Gulf States Utilities (GSU), in anticipation of GL 88-20 being issued in early 1988, began evaluating the methods used to perform an IPE. GSU operates River Bend Station (RBS) which is a 2894 MW(t) Boiling Water Reactor Product Line 6 (BWR/6). The BWR/6 reactor vessel is housed in a MARK-III Containment. RBS is located approximately 25 miles north of Baton Rouge, Louisiana.

In early 1987, the Engineering Department located at the RBS site began evaluating methods to perform the Level I PRA. Since the IPE is divided into two parts, i.e., Level I PRA and Level II PRA, each part must be evaluated. Since it was clear that the Level II PRA methodology was still evolving only the Level I PRA methodology was considered. A system analysis was begun to determine the best way to perform the Level I PRA (i.e., front-end).

This systematic analysis consisted of four major sections. The first section identified the details of the problem to be solved. For the case presented in this paper, the problem was to satisfy the NRC's Severe Accident Policy. It should be noted that this system analysis was performed prior to the NRC issuing GL 88-20, therefore no clear guidance was available on how the safety analysis should be performed.

The second section consists of identifying potential methods that could be considered in performing a Level I PRA which is also commonly referred to as a probabilistic safety assessment (PSA). The Level I PRA methods identified were 1) the detailed Level I PRA which models all important systems to the component level and has significant human reliability analysis (HRA) performed, 2) the focused-PRA which is documented in NUREG/CR-4550<sup>1</sup> and is used in the reference plant (Grand Gulf Nuclear Plant) PSA. NUREG/CR-4550 methods have been referred to as a focused-PRA because only the most important systems are modeled to the component level and the other systems are modeled in a more approximate way, and 3) the IDCOR-IPE methodology<sup>2</sup> which was developed to provide a simple approach in determining plant specific risk based on comparing plant specific data to a reference plant and making adjustments to the core damage frequency (CDF), accordingly.

The third section of the system analysis provided a detailed cost evaluation of each alternative and the fourth section identifies the advantages and disadvantages of each alternative. The fourth section also provides a recommendation based on the results of Section 3.

In reviewing the results from this system analysis and reviewing bids from several vendors, it was decided to perform the Level I PRA using the focused-PRA approach.

Science Applications International Corporation (SAIC) was selected as the contractor to provide technical support to GSU's in-house staff. A full time SAIC technical advisor was located at RBS site and two full time equivalent GSU engineers were assigned to work with SAIC engineers in performing the Level I PRA.

For the RBS specific Level I PRA, 23 system notebooks have been compiled, all the necessary event trees and fault trees have been developed, and all the fault trees have been integrated to form the plant model. Additionally, the necessary component and human reliability data have been assembled. Engineering personnel are currently quantifying the accident sequences identified in the event trees. Although the RBS Level I PRA is not complete, portions have been utilized to perform safety evaluations in support of plant operations.

A diagram of some event trees and fault trees are shown in Figure 1. As shown, non-safety related support systems are important in determining the CDF of a nuclear plant.

GSU is finalizing the Level I PRA and should be complete with this effort in the last quarter of 1989. Other factors must be integrated into the Level I PRA such as internal flooding. In addition, NUREG-1330 has given guidance on performing the Level II PRA and this effort will begin in late 1989.

Other NRC requirements are expected in 1989. These requirements are the evaluation of seismic and fire risk and the integration of accident management into utility operations. These additional requirements may cost the utility up to 3 to 5 million dollars in addition spending in the next five years.

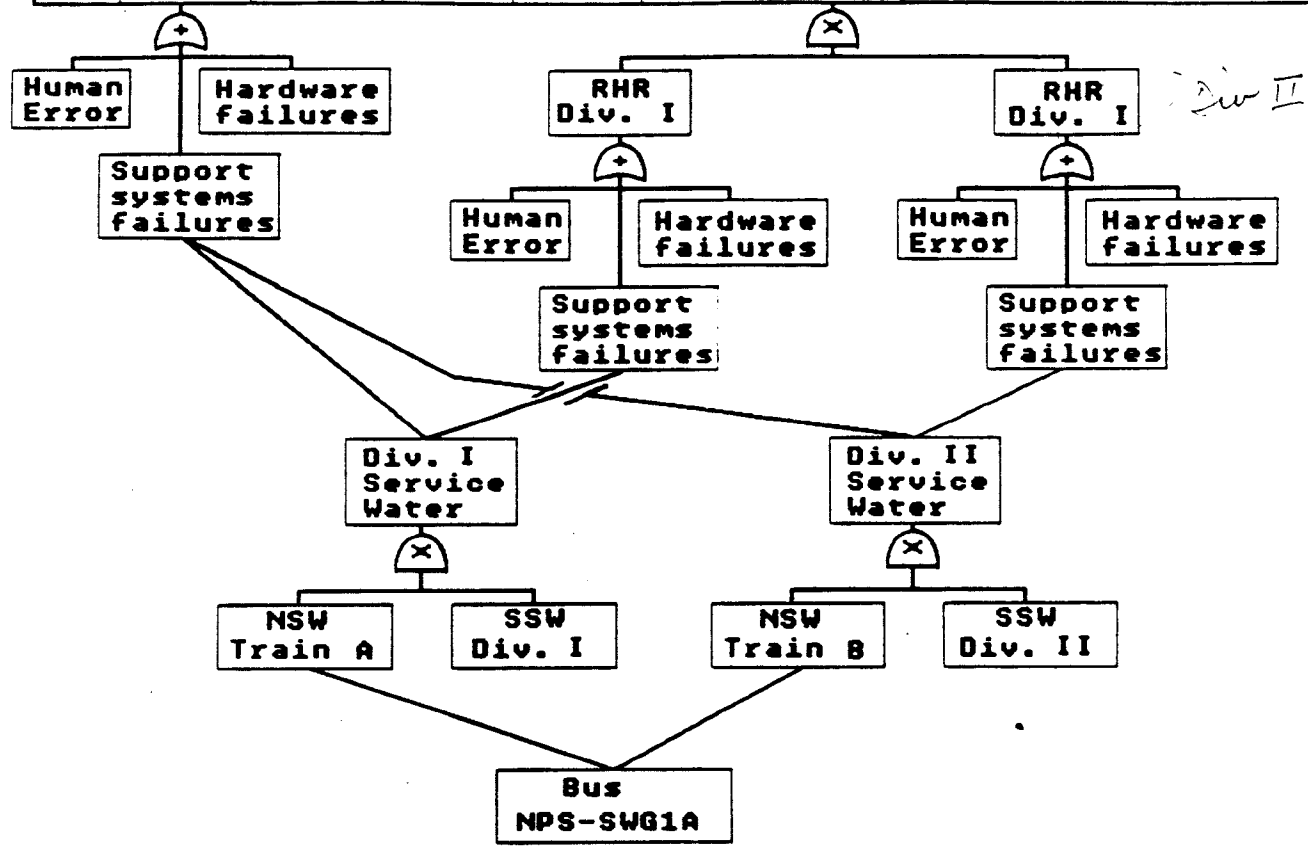
The next few years will provide much insight to utilities determining potential safety weaknesses and strengths of their nuclear power plants. It is hoped by the utilities that the NRC mandated requirements can also be used as tools by the plants to obtain relaxed technical specification requirements in testing and maintenance activities. It is only logical that if a tool can identify potential safety weaknesses, the same tool can also identify redundant requirements that only add complexity to the plant and does not provide additional safety margin.

## References

1. Drouin, M. T., et. al., "Analysis of Core Damage Frequency From Internal Event: Methodology Guidelines", NUREG/CR-4550, Vol. 1, SAND86-2084, Sandia National Laboratories, Albuquerque, NM, and Livermore, CA, September 1987.
2. Industry Degraded Core Rulemaking (IDCOR) Program, "Individual Plant Evaluation Methodology For LWRs", April, 1987.

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							6	CM	—
							7	OK	—
							8	CM	—
							9	CM	—
							10	CM	—
							11	CM	—
							12	CM	—

EVENT TREES



FAULT TREES

**EVENT TREES** - "Map" System Responses Leading To Core Damage

**FAULT TREES** - Models the Ways a System can fail

## EVENT TREES AND FAULT TREES

Figure 1

PROCEEDINGS  
**Full Paper Cover Sheet**

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