USING PRA METHODS AT AN OPERATING NUCLEAR UTILITY

BY

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River Bend Station

St. Francisville, Louisiana

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RIVER BEND STATION

- June 1986 - Commercial Operation
- 2894 MWt/936 MWe
- Boiling Water Reactor - 6
- General Electric NSSS Vendor
- Mark III Containment
- Stone & Webster is Architect/Engineer
- Gulf States Utilities
- RBS supplies 15% of GSU's power
- RBS is approximately 75% of financial risk

NUCLEAR REGULATORY COMMISSION
SEVERE ACCIDENT POLICY

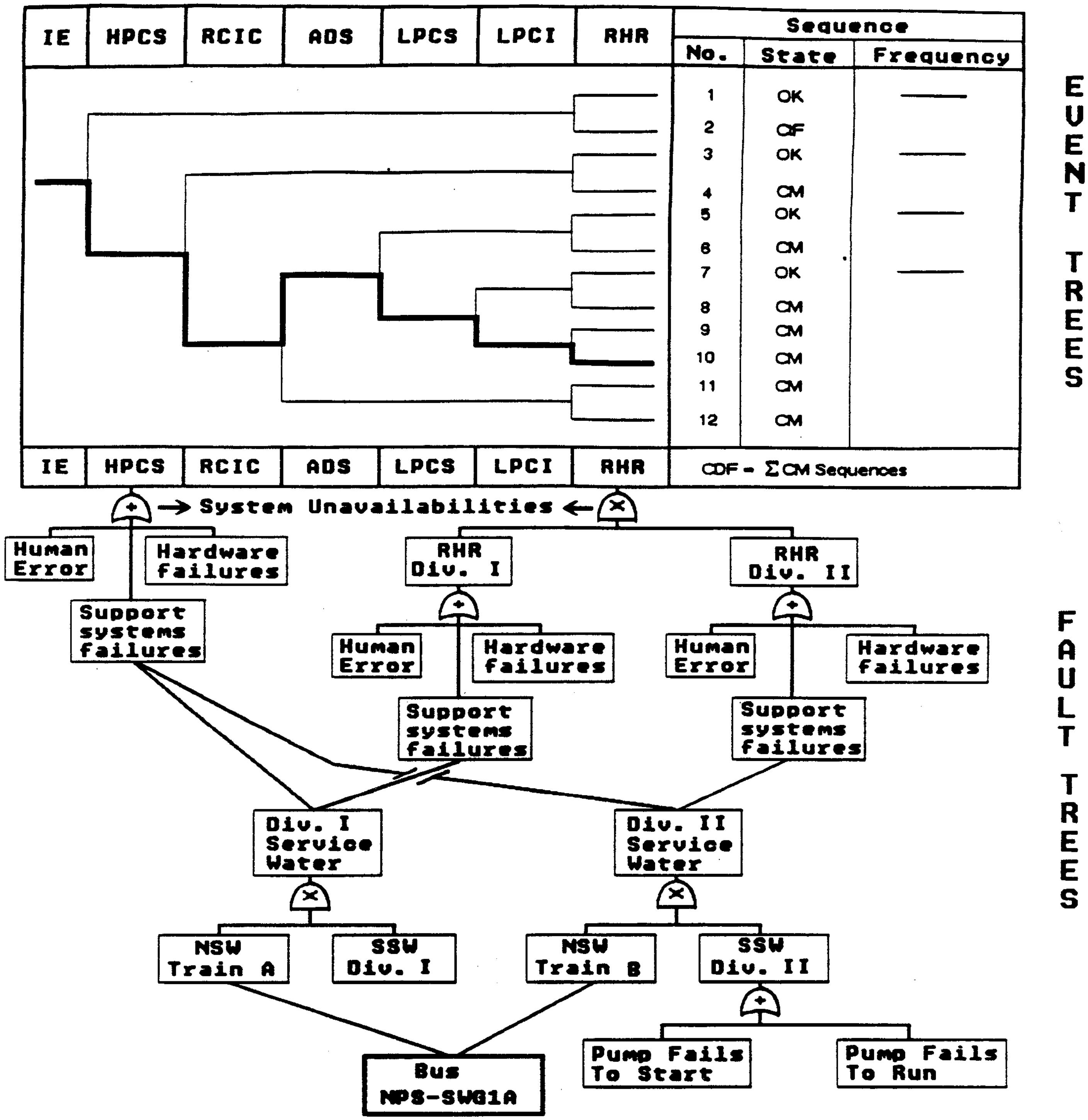
- Generic Letter on Individual Plant Examination
- Probabilistic Risk Assessment (PRA)
- Resolution of Safety Issues
- Accident Management
- Rational resolution of regulatory issues
- Improved procedures
- Closure of Severe Accident Policy

ELEMENTS OF PRA EVALUATION

- 19 System Fault Trees
  (e.g., Feedwater System, Reactor Protection System, etc.)
- 13 Initiating Events
  (e.g., Loss of Coolant Accident, Loss of Offsite Power, etc.)
- 250 Operating and Maintenance procedures (STPs, EOPs, AOPs, ARPs)
- Human Reliability Analysis Input
- 150 values for data input (IEEE-500, NUREG-4550, IREP-, etc.)
- 6 Common Mode Failure Data Group (MOVs, Motors, Diesel Generators, Fans, AOVs, Common Mode Miscalibration)
- Plant Specific Design Data
- Support System Dependencies

USES OF PRA

- Improve Operation and Maintenance decision making
- Prioritize design change requirements
- Enhanced awareness of utility personnel
- Improvement of normal and emergency operating and maintenance procedures
- Maintain safety margin during shutdown
- Support Waiver of Compliance and Technical Specification Change



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

FAULT TREES - Models the Ways a System can fail

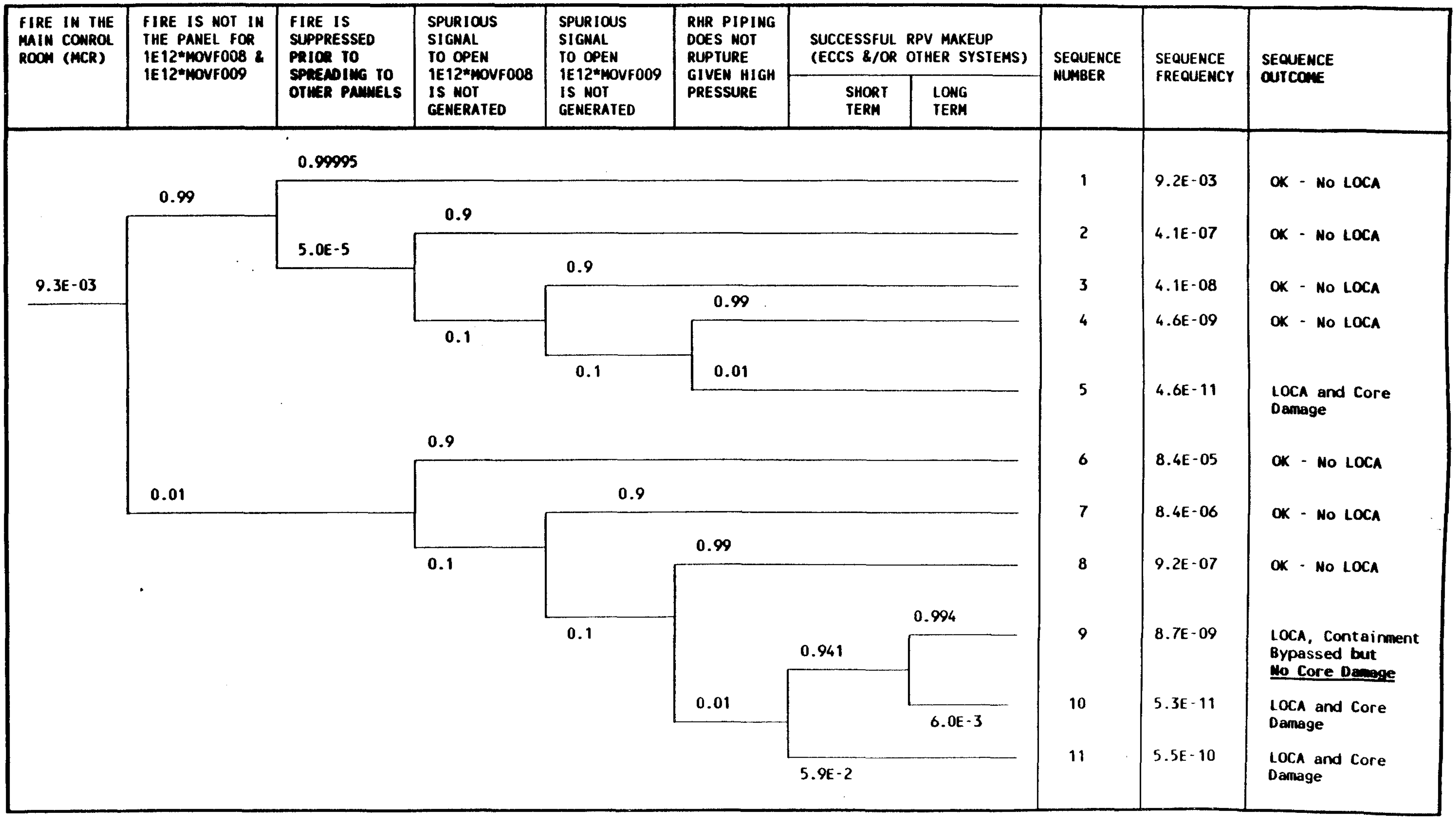
EVENT TREES AND FAULT TREES

Figure 1

EXAMPLES OF USE

- Waiver of Compliance on mode change with RCIC inoperable
- Ranking of MOVs by risk
- Evaluation of 54-month instrument calibration interval
- Safety impact of FHA valves not de-energized
- Safety impact of not having preferred station transfer

EVENT TREE FOR A INTERFACING SYSTEM LOCA
DUE TO A FIRE IN THE MAIN CONTROL ROOM



PROBABILITY OF PIPE RUPTURE = 1.0×10^{-2}

FOR ECCS - SHORT TERM ASSUMED THAT LPCS IS FAILED, ONLY HPCS CAN RESPOND TO BREAK

TOTAL CORE DAMAGE FREQUENCY DUE TO FIRES IN MCR = 6.5×10^{-10} Per Reactor-Year

COMPARISON OF RIVER BEND TRANSIENT SEQUENCES

NORMAL CONFIGURATION W/. PREFERRED E & B AVAILABLE

	6/YR	6.0xE-04/YR	1.6xE-06/YR	1.6xE-08/YR
100% POWER	SCRAM OCCURS FREQUENCY = 6/YR RESPONSE: LOSS OF NORMAL XFORMERS OPERATOR ACTIONS: AOP-1 REACTOR SCRAM AOP-2 MAIN TURB TRIP MUST ADDRESS CAUSE OF SCRAM STATUS: NORMAL SCRAM	LOSS OF PREFERRED-E AND FEEDWATER PROB = 1.0xE-04 RESPONSE: LOSS OF NPS-A LOSS OF FEEDWATER DEGRADED NSW DEGRADED CONDENSATE DEGRADED INSTR AIR DEGRADED RECIRC AUTO START SSW OPERATION ACTIONS: AOP-3 AUTO ISOLATIONS AOP-6 COND/FW FAILURES AOP-10 LOSS OF ONE RPS AOP-11 LOSS OF RPCCW AOP-24 DECREASE IN RECIRC STATUS: N.O.U.E.	LOSS OF SSW PROB = 3.2xE-03 RESPONSE: LOSS OF ECCS (4-16 HOURS) OPERATOR ACTIONS: AOP-16 LOSS OF SSW STATUS: ALERT POTENTIAL UPGRADE TO S.A.E. HPCS AND RCIC FOR MAKEUP	OPERATOR RECOVERY ACTIONS FAIL PROB = 1xE-02 POTENTIAL RECOVERY ACTIONS: RESTORATION OF ROOM COOLING REALIGNMENT OF SSW CROSS TIE NPS A&B BUSES RECOVERY OF HEAT SINK STATUS: HPCS AND RCIC FAIL ON ROOM COOLING SITE AREA EMERGENCY

CONFIGURATION W/O NORMAL-B TRANSFORMER

	6/YR	2.0xE-02/YR	7.0xE-05/YR	7.0xE-07/YR	7.0xE-08/YR
100% POWER	SCRAM OCCURS FREQUENCY = 6/YR RESPONSE: LOSS OF NPS-B DEGRADED FEEDWATER DEGRADED CONDENSATE DEGRADED INSTR AIR RECIRC PUMP TRIP DEGRADED NSW OPERATOR ACTIONS: AOP-1 REACTOR SCRAM AOP-2 MAIN TURB TRIP AOP-3 AUTO ISOLATIONS AOP-6 COND/FW FAILURES AOP-10 LOSS OF ONE RPS AOP-24 DECREASE IN RECIRC STATUS: MORE SEVERE THAN NORMAL SCRAM MORE SEVERE THAN IAS SCRAM	LOSS OF PREFERRED-E PROB = 3.5xE-03 RESPONSE: LOSS OF NPS-A LOSS OF FEEDWATER LOSS OF CONDENSATE LOSS OF INSTR AIR LOSS OF RECIRC LOSS OF NSW LOSS OF CB CHILL WATER OPERATOR ACTIONS AOP-5 LOSS OF COND VAC AOP-9 LOSS OF NSW AOP-11 LOSS OF RPCCW STATUS: N.O.U.E. POTENTIAL UPGRADE TO ALERT MORE SEVERE THAN IAS SCRAM	LOSS OF SSW PROB = 3.2xE-03 RESPONSE: LOSS OF ECCS (4-16 HOURS) OPERATOR ACTIONS: AOP-16 LOSS OF SSW STATUS: ALERT POTENTIAL UPGRADE TO S.A.E. HPCS AND RCIC FOR MAKEUP CORE DAMAGE W/O RECOVERY	OPERATOR RECOVERY ACTIONS FAIL PROB = 1xE-02 POTENTIAL RECOVERY ACTIONS: RESTORATION OF ROOM COOLING REALIGNMENT OF SSW STATUS: HPCS AND RCIC FAIL ON LOSS OF RM COOLING CORE DAMAGE W/ RECOVERY SITE AREA EMERGENCY	SPECIAL RECOVERY ACTIONS FAIL PROB = 0.1 RECOVERY OF NSW VIA GRANT SUB TIE-IN RECOVERY THROUGH FANCY POINT (LOW IMPACT)

CONCLUSIONS

- PRA is an extremely important tool to evaluate plant safety
- PRA is the only practical way to quantify safety
- PRA can provide assurances to the NRC that the plant is under safety goal of 10^{-4}
- PRA can save significant resources to help management make smart decisions
- Provides evidence to the NRC that utility personnel know how to operate the plant safely

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INTRODUCTION

Since the Three Mile Island-2 (TMI-2) accident in 1979, severe accident issues have permeated Nuclear Regulatory Commission (NRC) thinking in regulating nuclear power station construction, design, operation, and maintenance activities. Subsequent to the TMI-2 accident, the NRC formulated a Severe Accident Policy.

The Severe Accident Policy is a guide to regulatory decision-making. The policy provides general procedures for staff approval of items related to severe accidents. Even though the NRC has stated that it believes existing nuclear power plants pose no undue risk to the health and safety of the public, the Severe Accident Policy statement indicates that all plants will be required to perform a probabilistic safety analysis (PSA) to verify that this conclusion is true and to identify any potential outliers that might be plant specific. Plant specific safety evaluations can be accomplished through the use of Probabilistic Risk Assessment (PRA) methods. The NRC has recognized the value of PRA in assuring the safety of plant design and operation. The NRC notes that PRA has been useful in identifying risk outliers. The NRC also acknowledges that many of the modifications to address these outliers (or vulnerable areas) involve procedural changes or low cost design changes.

A full scope PRA is composed of three segments. The first segment is the Level I PRA and is considered a PSA leading to the determination of core damage frequency (CDF). The CDF provides a number which represents how often the reactor core can be damaged per reactor year of operation. For example, a CDF equal to 10^{-6} means that the reactor core could be damaged once in every million years. The CDF provides analytical evidence based on important initiating events (e.g., loss of coolant accidents (LOCAs) and potential transients) and the availability of equipment important to mitigating core damage. The second segment of the PRA is the Level II PRA which provides analytical evidence of the containment failure probability, which may or may not require reactor vessel melt through. The third segment of the PRA is the offsite consequences to the general public subsequent to the failure of the containment. The PRA method of analysis is the most practical way of quantifying nuclear reactor safety with respect to the use and operation of procedures and equipment to mitigate the consequences of a severe accident.

In 1988, the NRC issued Generic Letter (GL) 88-20 and NUREG-1335 which required each utility owning a nuclear power facility to complete a systematic evaluation to determine CDF and containment failure frequency. GL 88-20 required all plants to perform an NRC acceptable evaluation that contained most of the full scope Level I and II PRA. In 1987 Gulf States Utilities (GSU), in anticipation of GL 88-20, began work on a focused-PRA[1] of River Bend Station (RBS), a 2894-MW (thermal) General Electric boiling water reactor product line 6 (BWR/6) operated by GSU. The BWR/6 reactor vessel is housed in a GE MARK-III containment. RBS is located approximately 25 miles north of Baton Rouge, Louisiana.

Since the PSA is the first segment of the PRA, GSU began using PSA type evaluations in 1987 to support plant operation, maintenance activities, nuclear licensing regulatory issues, and engineering decision-making. The RBS plant specific PSA was developed using methods provided by NRC contractors[2].

DESCRIPTION OF PRA METHODS

GSU believes emphasis should be placed on mitigating core damage, therefore the figure of merit used in assessing safety significance is CDF. This can be determined by using the results of the PSA (i.e., Level I PRA). The PSA is comprised of an extremely detailed model of 23 major systems that are important to plant safety with a large number of these systems modeled to the component level. The modeling methodology employed by GSU is documented in Reference 2. An example of a typical sequence model is shown in Figure 1. The sequence model is composed of event trees which identify the initiating events and systems called upon to mitigate the consequences of the event. As seen in Figure 1, the event can progress depending on the availability of mitigating systems to an "OK" status (i.e., no core damage) or to a "CM" status (i.e., core melt or core damage). The fault trees provide system unavailability information to the event tree models as shown in Figure 1. The fault trees consist of components and other contributing factors that model the system functionability. Also included are factors that integrate human error and various types of equipment failure probability. As one can see in Figure 1, interactions with other support systems are also modeled. With this detailed model one can see how various parameters can be changed in the model to determine safety significance.

For example, instead of the normal service water pump being unavailable based on normal availability of the pump, the pump may be out of service based on maintenance activities. This would change the unavailability of the service water pump from .005 to 1.0 and therefore propagate through the analytical model calculation to ultimately impact the CDF. This same technique can be used to assess human reliability assumptions, changes to maintenance practices, operational changes, and design changes. GSU has used this concept to assess the safety impact on many decisions made at RBS.

As the PSA method of assessing plant safety became known to RBS plant staff, more requests to evaluate changes to the plant and to resolve regulatory questions were received. These types of assessments provided valuable insights to plant management in addition to beneficial cost reduction by eliminating or reducing work activities on equipment that had little impact on plant safety. Also these insights were used to convince regulatory agencies that some required activities could be delayed without impacting plant safety.

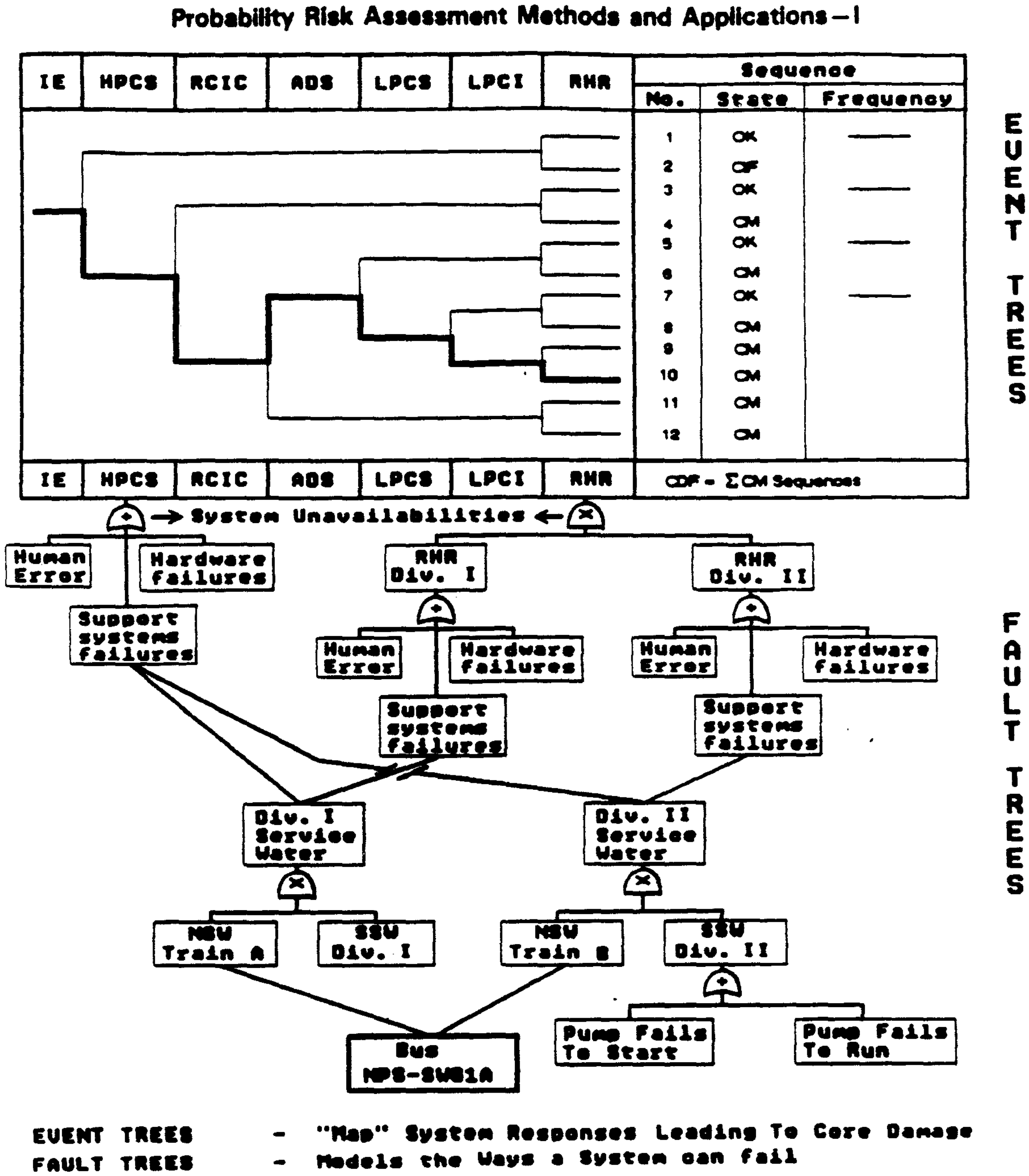

Fig. 1. Event trees and fault trees.

TABLE 1
PRA PLANT SUPPORT 1987 to PRESENT

	CUSTOMER	DOCUMENT	OUTAGE
1) Operations with ADS inoperable due to SVV compressors	Licensing/Plant Staff	Direct Request	
2) Drywell hydrogen igniter safety assessment	Licensing/EQ	LER 90-048	
3) Waiver of compliance on drywell airlock doors	Licensing/Plant Staff	T/S 3.6.2.3	RF3
4) Waiver of compliance on Mode change with RCIC inoperable	Licensing/Plant Staff	T/S 3.0.4 & 4.7.3b	RF3
5) Evaluation of HPCS Rosemount transmitters	Licensing/Plant Staff	CR 90-1103	RF3
6) Ranking of MOV's by risk	Design Engineering	NRC GL 89-10	RF3
7) Estimate of the probability of SSE occurring in a 30 day period in support of RF3 control rod storage plans	Plant Staff	Direct Request	RF3
8) Probability of SSE or OBE	Licensing	TSI (OPDRV)	RF3
9) Review of BWROG safety assessment on MOV isolation	Design Engineering	NRC GL 89-10	RF3
10) Standby service water system single active failure analysis	Licensing	NRC GL 89-13	RF3
11) Reliability study for Olin 450 psig	GSU Industrial Marketing	Direct Request	
12) Asea Brown Boveri circuit breaker failures	Operations	EEAR 89-R0161	
13) Impact of a short on DIV III bus while testing DIV III diesel generator	Design Engineering	Direct Request	
14) Feedwater control - high level trip power supply	Design Engineering	Direct Request	
15) 24 hour extension of LCO due to 1HVC*AHU2A	Licensing	LCO	
16) Review of Scram 90-02	Operations	Direct Request	
17) Plant operations without auxiliary boiler	Design Engineering	Direct Request	
18) Evaluation of 54 month instrument calibration interval	Maintenance	CR 89-1265	
19) Safety evaluation of proposed alternate electrical distribution alignment	Design Engineering	Direct Request	
20) Analysis of the safety impact of having valves listed in the FHA as de-energized, energized	Design Engineering	Direct Request	
21) Topaz inverter analysis	Design Engineering	Direct Request	
22) Temperature of a conduit near 1G33*MOVF040	Design Engineering	Direct Request	
23) NRC Augmented Inspection Team on interfacing system LOCA	Senior Management	CR 90-0116	
24) Comparison of RETRAN model vs. simulator model for ADS actuation at 100% power	Operations/Training	CR 90-0116	
25) PSA of alternate shutdown cooling during mid-cycle 3 outage	Outage Management	EEAR 89-E0218	
26) Effect of having a preferred station transformer out of service on core damage frequency	Senior Management	Direct Request	RF2
27) Probability of Anticipated Transient Without Scram (ATWS) Events Leading to Core Damage	Design Engineering	Direct Request	
28) Risk/Benefit analysis on turbine stop/control valve testing with bypass	Design Engineering	MR 89-0046	
29) Single failure scram analysis	ISEG	Direct Request	

USES OF PRA

Table 1 provides a summary of uses for the PRA at RBS. The first column in Table 1 provides a summary description of the task, the second column identifies the customer (i.e., major groups that are supported at RBS), the third column identifies the document or source that requested the work to be performed and the fourth column identifies outage related work. The customer can be divided into three major support areas (i.e., regulatory (6 items), plant staff (11 items) and engineering (12 items)). As the RBS PRA becomes more mature, more integration into maintenance and technical specification support will take place.

The regulatory evaluations are typically performed to satisfy NRC questions and to support the Technical Specifications questions.

EXAMPLES OF USES

During a recent plant startup, it was determined that the reactor core isolation cooling (RCIC) system was inoperable. Without RCIC operable, the plant was prohibited by Technical Specifications from entering the RUN mode. Therefore, the plant startup was placed on hold until a waiver of compliance could be obtained from the NRC to allow entrance to RUN mode without RCIC operable. As part of the technical justification for this waiver of compliance, a PSA was performed assuming RCIC inoperable. With RCIC operable, total CDF was 2.07×10^{-6}. With RCIC inoperable, total CDF was increased to 9.6×10^{-6}, or by a factor of 4.6. However, this was still an order of magnitude below the NRC safety goal of 1.0×10^{-4} for core damage events. Also, this change is within the uncertainty of PRA techniques. This PSA

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FIGURE 2

evaluation helped convince the NRC to grant the requested waiver of compliance, which allowed plant startup to continue. Without this waiver, the plant could not start-up until the RCIC system became operable. Although the cost benefit would depend on the length of time required to have an operable RCIC system, based on a $180,000 per day replacement power cost the cost savings are considered significant.

As part of River Bend Station's response to NRC Generic Letter (GL) 89-10, the PRA group was asked to evaluate priority of motor-operated valves (MOVs) for inclusion in the MOV testing program. As a result, sixty (60) MOVs were determined to be most important relative to plant safety. A list of these valves was developed and provided to Design Engineering for their use in preparing the MOV testing schedule. The benefit in this evaluation allowed resources to be allocated only to the most important activities with respect to safety and allowed the deferment of activities that had a minimal impact on plant safety thereby optimizing allocation of scarce resources.

During normal operations, one train of the heating, ventilation, and air-conditioning (HVAC) serving safety-related switchgear and battery rooms was inoperable. Plant Staff anticipated the need for a 24-hour extension to the 7-day Technical Specification Limiting Condition for Operations (LCO) governing this situation. A PSA was performed to support the LCO extension request. This analysis demonstrated that the 24-hour extension increased CDF from 3.61×10^{-6} to 3.65×10^{-6}, or an increase of only 4×10^{-8} per year. Therefore, the LCO extension had no significant impact on plant safety. This LCO extension allowed the plant to continue to operate without a costly shutdown thereby saving significant resources for GSU.

Plant Engineering requested an evaluation of the potential plant maintenance impact of extending the calibration intervals to 54-months for a number of safety-related instruments. These instruments were divided into two groups; those that provide information or alarm functions only and those that provide signals for automatic action. PSAs were performed on the instruments controlling automatic actions to demonstrate that this extension would have no significant impact on the core damage frequency at River Bend Station. For the instruments providing information or alarms, operating procedures were reviewed to determine their impact, if any, on the core damage frequency. Based on this evaluation, plant maintenance requirements were reduced for 28 instruments without impacting plant safety. This provides a reduction in maintenance cost for the life of the plant and a significant cost savings for GSU in reduced maintenance manpower requirements.

During normal plant operations, several valves which were identified in the Fire Hazard Analysis (for Appendix R requirements) as being de-energized were discovered never to have been de-energized. Of the valves identified, one of the NRC's major concerns was the potential for a fire-induced interfacing system LOCA in the residual heat removal (RHR) shutdown suction line. A PSA was performed to determine the safety significance of being in this configuration. This analysis demonstrated that the CDF due to a fire-induced interfacing system LOCA was approximately 3×10^{-9}. Based on the fact that this CDF was well below the safety goal of 10^{-4} for CDF and 10^{-6} for a large release, it was concluded that the event had an insignificant impact on the health and safety of the public. This PSA helped convince the NRC that this event had little safety significance and helped eliminate a costly fine which could have been as much as $100,000.

At the end of Refueling Outage-2 (RF2), Engineering Analysis was requested to determine the impact on safety due to operating with one non-safety-related transformer out of service. Although this transformer was not required to be in compliance with Technical Specifications, a PSA indicated plant operation without this transformer could increase the core damage frequency significantly. During this analysis, a comparison between the progression of the accident, under both the normal and current transformer configurations, to the required operator response was developed. This comparison is shown in Figure 2. This method of presentation allowed the differences to be easily noted. Based on this PSA, plant start-up was delayed approximately two weeks while a replacement transformer was installed. Although plant start-up was allowed by Technical Specifications, GSU felt the prudent action was to delay start-up to minimize the chances for adverse operation of the plant. In this case, an additional cost for continued outage was incurred by GSU, but GSU avoided a high risk situation with a qualified analysis.

CONCLUSION

The PRA at River Bend Station was initially developed to address the requirements of NRC Generic Letter 88-20. However, in addition to fulfilling this regulatory role, the River Bend PRA has been utilized to support plant evolutions. At River Bend, the PRA represents an analytical tool capable of evaluating changes to plant design, operations, and maintenance. Supporting requests for waivers of compliance and LCO extensions, quantifying the safety significance of components, and justifying extensions of calibration intervals are a few examples of the changes which PSAs can evaluate. Such changes can have either positive or negative impacts on plant safety and reliability. Integration of PRA techniques into day to day decision-making at the plant can lead to significant resource savings while ensuring that margins of safety are preserved.

REFERENCES

1. Miller, J.S. and Cathey, N.G., "Implementation of an Individual Plant Examination at a Nuclear Utility." Paper presented at the American Nuclear Society 1989 Winter Meeting, San Francisco, Ca., November 26-30, 1989.

2. Drouin, M. T., et. al., <u>Analysis of Core Damage Frequency from Internal Event: Methodology Guidelines</u>, NUREG/CR-4550, Vol. 1, SAND86-2084, Sandia National Labs., Albuquerque, New Mexico (September 1987).