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Importance Ranking Based on Aging Considerations of Components Included in Probabilistic Risk Assessments

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ABSTRACT

This report presents a method for focusing additional research on aging phenomena that affects nuclear power plant components. Specifically, the method ranks components using a risk aging sensitivity measure that describes the change in risk due to changes in component failure rate. Describing the aging phenomena and the resulting time-dependent component failure rate changes is beyond the scope of this study.

The applications use average component unavailability equations currently employed in PRAs to calculate the risk aging sensitivity. A more exact calculation is possible by using unavailability equations that include the time-dependent characteristics of component failure rates; however, these time-dependent characteristics are not well-known. The risk aging sensitivity measure presented here is, therefore, segregated from these time-dependent effects and addresses only the time-independent portion of aging phenomena. The results identify the component types that show the most potential for risk change due to aging phenomena. Future research on the time-dependent portion of aging phenomena for these component types is needed to completely describe the risk impact due to component aging.

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EXECUTIVE SUMMARY

This study utilizes existing probabilistic risk assessments (PRAs) to gain insights about the relationships between aging of nuclear power plant components and public risk. A method is developed and applied for determining the potential risk significance of aging effects. This method is based on determining the sensitivity of risk to increases in component failure rates. The partial derivative of the core melt frequency with respect to the failure rate of a specific component is the risk aging sensitivity measure used. Those components having the highest sensitivity have the potential for causing the greatest change in risk if their failure rates increase due to aging or service wear.

PRA has shown these [illegible] most important as indicators of bad safety performance

The results of the analysis indicate the most risk significant components at a plant depend on a number of factors including plant system design, testing, and maintenance intervals and operating procedures. Based upon the three PRAs analyzed (Oconee, Calvert Cliffs and Grand Gulf) many of the potentially most risk significant components are in the auxiliary feedwater system, the reactor protection system and the service water systems. Pumps, check valves, motor operated valves, circuit breakers, and actuating circuits are the component types that have the most potential risk impact based on the aging sensitivity measure.

i.e. components [illegible] which [illegible] have the greatest impact on safety acceptance of license renewal.

The components listed have sometimes for both INEL and ANL but are all active except maybe for check valves.

The results of this study are intended to provide guidance for this selection of components for further study in the aging program and as a guide toward prioritizing resources. The results presented are subject to several assumptions and limitations. The risk aging sensitivity measure used does not describe the time-dependent behavior of the failure rate. In addition no assumptions are made about which components are most susceptible to aging processes. Other key limitations of this study are the limited number of plants analyzed and limited scope of the PRAs performed for these plants. Only the components which appeared in the PRAs were considered in detail. Components not analyzed in the PRAs or components assumed to have negligible failure rates can be important to risk if their failure rates increase substantially. The study suggests future research activities which would address many of these limitations.

these are two things we can ultimately furnish info on!

The output from this study can be combined with other studies (data, analytical or experimental) that identify the components that are most susceptible to aging mechanisms. The combination of identification of risk significance and aging susceptibility will provide a good basis for effectively focusing resources.

1. INTRODUCTION

The overall goals of the Nuclear Plant Aging Research (NPAR) Programs are:

- To identify electrical and mechanical component aging and service wear effects likely to impair plant safety.
- To identify methods of inspection and surveillance of electrical and mechanical components that will be effective in detecting significant aging and service wear effects prior to loss of safety function so that proper maintenance and timely repair or replacement can be implemented.
- To identify and recommend acceptable maintenance practices which can be undertaken to mitigate the effects of aging and to diminish the rate and extent of degradation caused by aging and service wear.

The NPAR program is being performed by the NRC Office of Nuclear Regulatory Research and its contractors. Several specific research activities have been established to achieve the overall program goals.

ours also?

The objectives of this study concern only the first goal. Our objective is to identify components in nuclear power plants that adversely affect risk if aging processes decrease component reliability or degrade performance characteristics. This objective does not include identifying specific aging processes or describing aging effects on component failure rates.

The approach taken in this study uses the results of existing probabilistic risk analyses (PRAs) to gain insights about the relationship between risk and component aging or wear-out. PRAs performed to date do not explicitly model risk as a function of time, but calculate an average risk level. This report defines a risk importance measure that measures the sensitivity of risk to changes in a component failure rate. This measure is the partial derivative of the core melt frequency with respect to the failure rate of a specific component. Those components that have the highest sensitivity have the potential for causing the greatest change in risk if their failure rates increase due to aging or service wear. The development of the aging sensitivity measure is described more fully in Section 2.0. Results of application of the aging sensitivity measure to components in selected PRAs are presented in Section 3.0.

The output from this study can be combined with other studies (data, analytical or experimental) that identify the components that are most susceptible to aging mechanisms. The combination of identification of risk significance and aging susceptibility will provide a good basis for effectively focusing resources. Section 4.0 presents the conclusions and recommendations of this study.

2. RISK IMPACT OF COMPONENT AGING

This section develops the aging sensitivity measure from the risk equations of a PRA. Some background information regarding PRAs is briefly reviewed to put the study in context. The second part of this section discusses the potential impacts of component aging on risk. The third subsection presents the aging sensitivity measure.

2.1 Risk Analysis

2.1.1 Background

PRAs are performed in order to assess the risk of nuclear power plants and to identify the key contributors to that risk. A number of insights developed from review of WASH-1400 (1) and other past PRAs are useful to focus aging related research.

The Reactor Safety Study (WASH-1400) was the first comprehensive study of the risk due to the operation of nuclear power plants. This study shows that the risk to the public from normal operation and routine releases is minimal. The risk is dominated by low probability, high consequence events where large amounts of radioactivity are released. In order for large amounts of radioactivity to be released, substantial fractions of the reactor core must melt. From a risk significance viewpoint, the aging processes of concern are those that could potentially affect the likelihood of core melt or affect the systems that mitigate the consequences of core melt.

2.1.2 Overview of PRAs

PRAs are a method to mathematically estimate the likelihood and the consequences of potential accidents at nuclear power plants. In the process of performing a PRA, the potential accident initiators (LOCAs, transients, loss-of-offsite power, etc.) are identified and their likelihood quantified. The safety systems and their support systems that must function to safely shut down the reactor are then identified for each initiator. The safety systems and their support systems are modeled using event tree and fault tree methodology. The safety systems generally considered in a PRA are the reactor protection system, main and auxiliary feedwater systems, high pressure and low pressure injection systems, residual heat removal systems, containment sprays, containment coolers, and accumulators. Support systems include electric power, service water, and engineered safety feature actuation systems. Operator actions are also included in the models.

The event tree and fault tree model solutions determine the combinations of component failures that lead to a core melt for each of the initiators. The combination of an accident initiator and the system failures that result in core melt is referred to as an accident sequence. The combinations of individual component failures that cause the required systems to fail is referred to as a cutset.

The probability of each individual component being unavailable is referred to as its unavailability. The probability of the cutset is the product of the unavailability of the individual events. The frequency of an accident sequence can be approximated by the sum of all the cutsets that result in failures of the same set of safety systems. The overall plant risk is similarly approximated by the sum of the accident sequences, or equivalently, the sum of all the accident cutsets.

In addition, a probability of containment failure can be assigned to each accident sequence. In some PRAs, the consequences of accident sequences are evaluated in terms of man rem, fatalities, or economic impact.

2.1.3 Scope of PRAs

The scope of PRAs vary greatly. Some consider internal events only; others include seismic events, floods and fires, etc. The depth of the analyses of the systems and sequence consequences also varies considerably. The scope of the PRA, as well as the level of detail considered, limits the information that can be extracted from the analysis.

PRAs generally concentrate on finding the most risk significant components. In many cases passive components such as the containment building, the reactor vessel, and storage tanks are considered to have negligible failure rates and are omitted from the risk analyses. In most PRAs, wires and piping segments are considered to have failure rates that are negligible when compared to the motors and valves with which they are associated and are omitted from further analysis. However, the risk significance of a particular wire or piping segment can be inferred from the PRA by determining the effect of failure of the wire or pipe on the component to which it is connected.

2.1.4 Risk Equations

In risk analyses, risk is expressed as a combination of frequencies of initiating events, probabilities that safety systems are failed and consequences of the sequence. The risk from a single accident sequence cutset can be expressed:

$$R_c = F \cdot Q_i \cdot C \qquad (1)$$

where

R_c = risk associated with the cut set

F = initiator frequency

Q_i = probability the components of the cut set i are all failed

C = consequence of the cut set.

In the above equations, the initiator could be a plant transient or a loss-of-coolant accident (LOCA) and the probability the necessary safety systems are unavailable may depend on which initiator has occurred. The consequence term, C, is a measure of the expected consequences of the sequence given a core melt. In this report we are limiting the analysis by considering core melt frequency as the measure of risk and will drop the C from the equation.

The plant risk, R_p, is the sum of all the accident sequences and is therefore expressed:

$$R_p = \sum_{\substack{\text{all} \\ \text{cut} \\ \text{sets}}} R_c \tag{2}$$

2.1.5 Unavailability Equations

The term Q in Equation (1) is the probability of a specific set of components are failed and is expressed

$$Q_i = \prod_{j=1}^{K} q_j \tag{3}$$

where

q_j = unavailability of component j

K = number of components in cut set i.

The unavailability term, q_j, for each component is dependent on a number of factors including the type of component, the testing interval, the failure rate, the time it takes to repair the component, the time period in which the component undergoes scheduled maintenance, and the likelihood of human error that affects the component. The types of components considered in this study fall into two general categories: periodically tested components and continuously monitored components. The unavailability equations for each type are presented below.

2.1.5.1 Periodically Tested Components

The average unavailability of periodically tested components consists of five terms, and the formula is expressed as the following:

$$\bar{q}_S = \bar{q}_F + \bar{q}_T + \bar{q}_R + \bar{q}_M + \bar{q}_H \qquad (4)$$

where

$\bar{q}_S$ = total average unavailability of the periodically tested component

$\bar{q}_F$ = average unavailability contribution from failure occurrence during the test intervals

$\bar{q}_T$ = average unavailability contribution from test period

$\bar{q}_R$ = average unavailability contribution from repair of failure

$\bar{q}_M$ = average unavailability contribution from scheduled/unscheduled maintenance

$\bar{q}_H$ = average unavailability contribution from human error.

The average unavailability contributions given that the failure rates are constant are presented below:

$$\bar{q}_F = \lambda_S \, T/2 \qquad (5)$$

$$\bar{q}_T = q_o \frac{\tau}{T} \qquad (6)$$

$$\bar{q}_R = \lambda_S \, T_R \qquad (7)$$

$$\bar{q}_M = \frac{d_M}{T_M} \qquad (8)$$

$$\bar{q}_H = C \qquad (9)$$

where

λ_S= constant standby failure rate

T = interval between tests

q_o = override unavailability (the probability that the component is inoperable during the test)

τ = test duration time

T_R = repair duration time

d_M = average maintenance duration time

T_M = average interval between maintenance

C = human error probability.

Hence:

$$\overline{q}_s = \frac{\lambda s T}{2} + q_o \frac{\tau}{T} + \lambda_s T_R + \frac{d_M}{T_M} + C \quad \cdot \qquad (10)$$

For some components, such as manually operated valves, the failure rate (λ_s) is extremely small, and can be assumed negligible. The formula for these components becomes:

$$\overline{q}_s = q_o \frac{\tau}{T} + \frac{d_M}{T_M} + C \; . \qquad (11)$$

It should be noted that the negligible λ_s is for a specific failure mode.

2.1.5.2 Continuously Monitored Components

The average unavailability of this class of components is the proportion of time that the component is inoperable in a relatively long period of time. Again, with the assumption that the failure rate is constant, the formula for the average unavailability is given below:

$$\overline{q}_o = \frac{\lambda_o T_R}{1+\lambda o T_R} \; . \qquad (12)$$

Approximately

$$\overline{q}_o = \lambda_o T_R \qquad (13)$$

where

$\overline{q}_o$ = average unavailability of continuously monitored components

λ_o = constant operating failure rate

T_R = repair duration time.

2.2 Aging Analysis

In order to evaluate the risk significance of aging phenomena, it is necessary to define what is meant by aging phenomena. For our purposes, "aging phenomena" are phenomena that have one of the following two effects:

(1) Cause the failure rate of a component to increase as a function of time, or

(2) Cause a component that was designed to meet certain standards to degrade such that it no longer fulfills its design requirements.

2.2.1 Effect of Increases in Failure Rate

The first aging effect considered causes the failure rate of a component (or a set of components) to increase with time as the components age or wear out. Figure 1 shows a sample plot of the failure rate λ as a function of time for a typical component. This is the familiar "bathtub" curve common to many components. This curve has three distinct regions: (1) the burn-in period, (2) the period of normal operation (where the failure rate is essentially constant), and (3) the wear-out period. Aging phenomena occur in the wear-out period where the failure rate is increasing. The root cause of this increase in failure rate results from any of a number of aging ~~phenomena~~, fatigue or corrosion, for example. The increase in the failure rate with time can have two effects on risk:

mechanisms as given in our Table and highlighted in the individual GAFL assessments.

(1) The increase in failure rate increases the unavailability (decreases the reliability) of a component important to safety

(2) The increase in failure rate of certain components could cause an increase in initiator frequency. This effectively increases the number of times safety systems must operate and proportionally increases the risk.

An example of a component where the unavailability increases with time is a pump in the low pressure injection system of a PWR. Normally the pump is in the standby mode and is tested at regular intervals. If the failure rate is increasing with time (as in the wear-out region of Figure 1), the unavailability history may look like that of Figure 2. In this example, the test interval remains constant but the fraction of tests detecting failures is increasing as the component ages. The unavailability of that component, and therefore the risk associated with

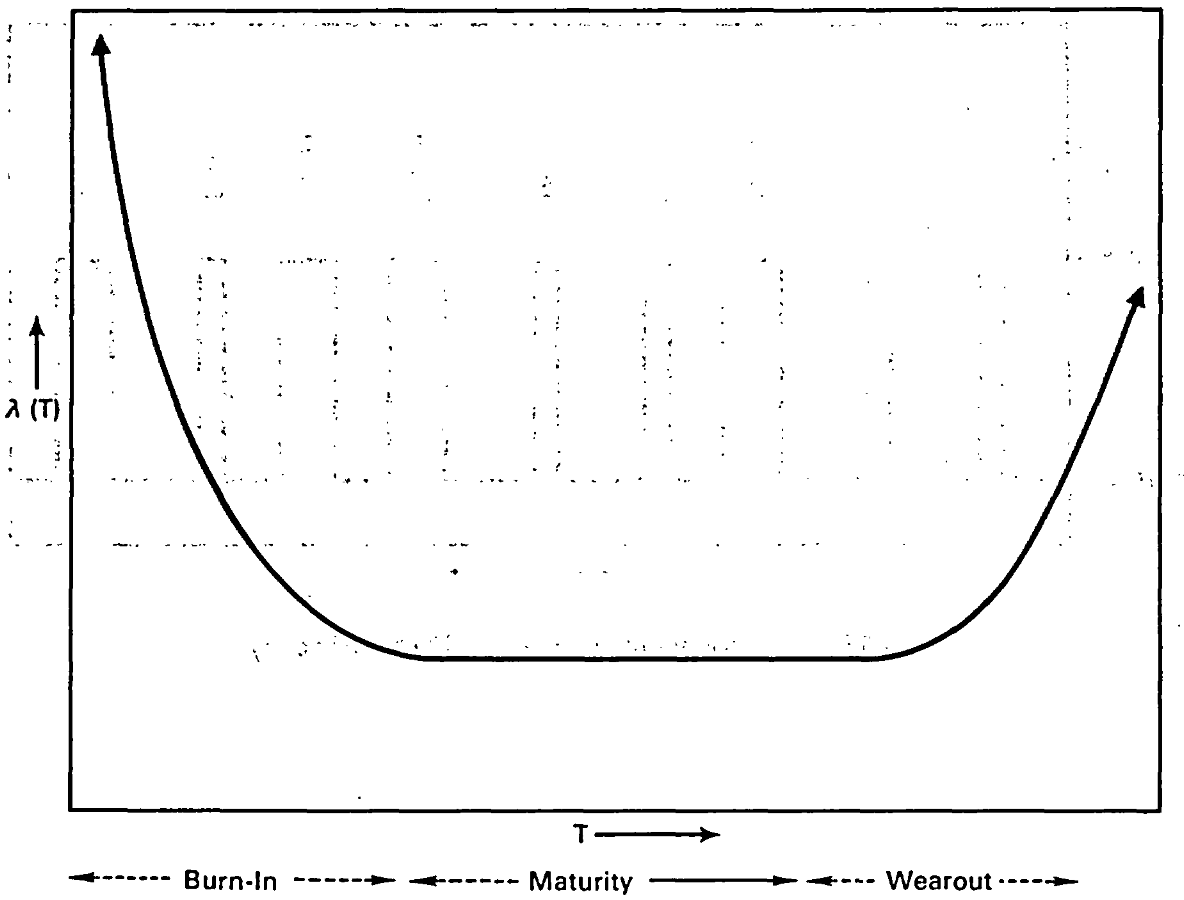

FIGURE 1. Example of a failure rate curve.

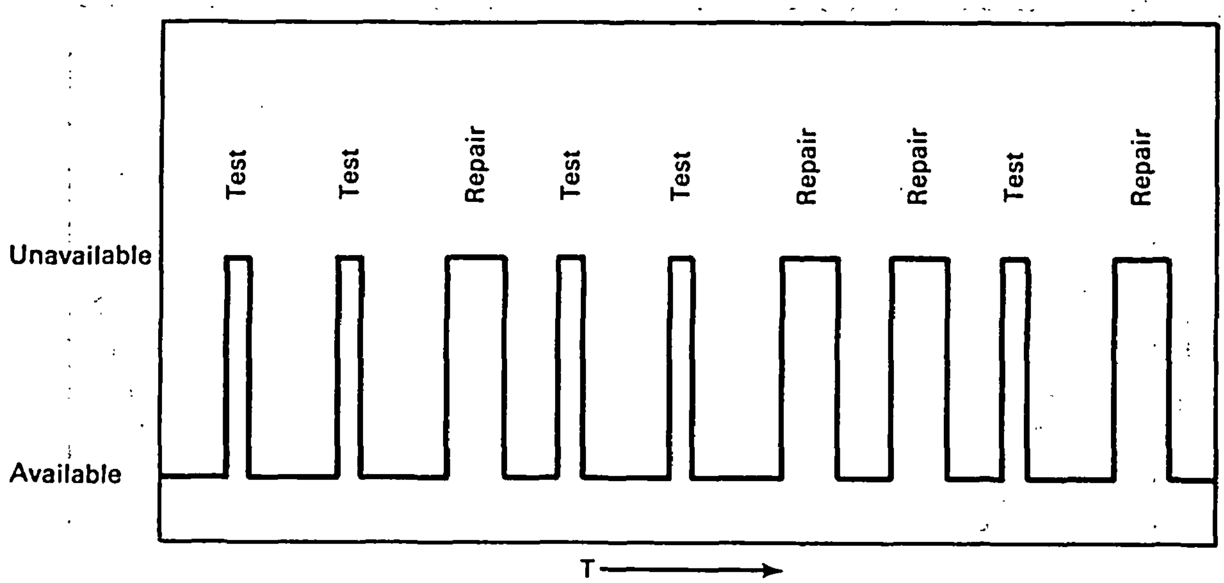

FIGURE 2. Component unavailability history.

that component, is increasing with time and may be substantially higher at the end of the period of interest than at the beginning.

supports component [illegible] of importance

An example of a component that could cause risk to increase by causing the initiator frequency to increase is a steam generator tube. If the failure rate of tubes is increasing, the likelihood of a steam generator tube rupture increases. Should this event occur, the necessary safety systems have to operate correctly to prevent core melt. Another example of components that increase risk by increasing the frequency of initiators is the reactor coolant system (RCS) piping. Also, components on the secondary side of the plant, such as the main feedwater pumps, whose failure rates increase with time have the effect of increasing the frequency of transient initiators and thus the risk.

2.2.2 Effect of Degraded Characteristics

The other type of aging phenomena that is of interest are processes that gradually degrade characteristics of the component. This could cause a component that is designed to meet certain design requirements to degrade such that it no longer fulfills its design requirements. Examples of this type of component are snubbers that lose their damping capacity as the fluid leaks through the seals or heat exchangers that lose heat transfer capacity as an oxidation layer is formed on the tubes. The reactor vessel can also be treated as this type of component since its pressure capacity decreases as a function of fluence. Determining the risk significance of this degradation is more complex than for components described in the last section since it generally involves combining a probabilistic load distribution with fragility curves and considering the impacts of the different failure modes. Current risk analyses generally consider all components to perform as designed under conditions of load and to be operating in accordance with design specifications. It will, therefore, be difficult to use PRAs directly to evaluate the risk significance of components of this type. However, bounding calculations can be performed.

2.3 Aging Sensitivity Measure

In order to characterize the risk impact of component aging and service wear effects, it is necessary to characterize the time dependent nature of the change in plant risk. That is,

$$I_A = \frac{\partial R}{\partial t} \tag{14}$$

where

I_A = risk impact due to aging

R = plant risk.

As defined in Section 2.1.4, plant risk is a function of component unavailability, q_j, and component unavailability is a function of component failure rate, λ. For the study of aging, the failure rate is a function of time, t. Taking advantage of the chain rule, changes in plant risk are expressed as

not this which is what needed. can we give this? with our current tables?

$$\frac{\partial R}{\partial t} = \frac{\partial R}{\partial q_j} \cdot \frac{\partial q_j}{\partial \lambda_j} \cdot \frac{\partial \lambda_j}{\partial t} \quad . \qquad (15)$$

This report only focuses on

The risk impact due to aging can now be separated into two distinct parts,

(1) The effects of changes in component failure rate on risk (the first two terms of the right hand side of Equation 15)

(2) The time-dependent effects of aging and service wear on the component failure rate (the third term of the right hand side of Equation 15).

This report concentrates on the first part, the change in risk due to changes in component failure rate. The second part, changes in the failure rate due to aging and service wear, is beyond the scope of this study and should be investigated through data evaluations, experimental studies, or additional analytical models. Section 4.0 describes how these two parts combine to describe risk impact due to aging.

We define the risk aging sensitivity to failure rate as

$$G_j = \frac{\partial R}{\partial \lambda_j} = \frac{\partial R}{\partial q_j} \cdot \frac{\partial q_j}{\partial \lambda_j} \quad , \qquad (16)$$

where the first term on the right hand side of Equation 16 is the partial derivative of risk with respect to component unavailability and the second term is the partial derivative of the component unavailability with respect to the component failure rate.

The first term, the partial derivative of risk with respect to component unavailability, can be shown to be equivalent to the Birnbaum measure.(2) This is a measure of the impact of a component failure on risk and can be computed by changing the unavailability of the component in the risk equation to unity and determining the change in risk. Vesely(3) et al have calculated values of the Birnbaum measure in recent work. The second term, the partial derivative of component unavailability with respect to component failure rate, is presented in Table 1. The expressions in Table 1 are derived from the component unavailability

Table 1. Rate of change of component unavailability with respect to failure rate.

Component Type	Average Unavailability	Rate of Change of Component Unavailability With Respect to Component Failure Rate
Periodically Tested Component	$\overline{q}_s = \frac{\lambda_s T}{2} + q_0 \frac{\tau}{T} + \lambda_s T_R + \frac{d_M}{T_M} + C$	$\frac{\partial \overline{q}_s}{\partial \lambda_s} = \frac{T}{2} + T_R$
Periodically Tested Component With Negligible Failure Rate	$\overline{q}_s = q_0 \frac{\tau}{T} + \frac{d_M}{T_M} + C$	$\frac{\partial \overline{q}_s}{\partial \lambda_s} = 0$
Continuously Monitored	$\overline{q}_0 = \lambda_0 T_R$	$\frac{\partial \overline{q}_0}{\partial \lambda_0} = T_R$

equations in Section 2.1.5. This second term is related to the time a component is unavailable when it is failed.

Table 1 also includes a risk aging sensitivity for components with negligible failure rates. This type of component unavailability is dominated by constant contributions, for example, human error, and represents an essentially time-independent unavailability. In this case the risk aging sensitivity factor is zero.

The risk aging sensitivity measure is used to rank components based on their potential for risk change. The measure makes no assumptions about the rate of component aging; the ranking results are valid only when all the components age at the same rate. Differences in aging rates between different component types is beyond the scope of this study and must be addressed in future research to describe the time-dependent behavior of component failure rates.

Section 3 presents the results obtained by applying the aging sensitivity measure to the components at selected plants.

3. APPLICATION OF THE RISK AGING SENSITIVITY MEASURE AT SELECTED PLANTS

In this section we present the results of risk aging sensitivity measure calculations for plants analyzed as part of the Reactor Safety Study Methodology Application Program (RSSMAP)[4,5,6]. These studies represent limited-scope PRAs in that they do not include external events and do not specifically include analysis of piping and wiring. The plants included in this analysis are two PWR's, (Oconee and Calvert Cliffs) and one BWR (Grand Gulf). Also included in this section are bounding calculations for three other components: a reactor vessel, steam generator tubes, and snubbers.

3.1 Component Boundaries and Failure Modes

The term "component" can be interpreted differently. In one sense, "components" can be considered individual pieces of hardware, e.g., a valve casing, a valve stem, wiring, etc. The "component" can also be considered as a functional unit such as a motor operated valve that consists of a number of component parts. Components as defined in most PRAs and in this report represent functional units. A motor operated valve for instance is interpreted as consisting of the valve, the motor operator, the circuit breaker, and the electrical cable and control circuitry specifically associated with the valve. A brief description of the component boundaries for each type of component is included in Table 2.

Frequently, components are subject to a number of different failure modes. For instance, motor operated valves could fail to function by several modes including: failure to open, failure to close, and gross leakage. Table 2 also includes the most important failure modes for each component type. These failure modes represent component functional failures and do not indicate the root cause of the failure or the failure mechanism. From an aging perspective, the time dependent processes that lead to a functional failure are of the most concern.

3.2 Results for Components at RSSMAP Plants

The risk aging sensitivity measure is calculated for individual components at each plant. The individual components are grouped by component type and also listed in order for each plant.

Table 2. Component boundaries.

Component	Boundary	Failure Modes of Concern
Pumps (Electric)	Includes pump, motor, and the control circuitry and electric power components specifically associated with the pump.(1)	• failure to start on demand • failure to run • gross leakage
Pumps (Turbine Driven)	Includes pump, turbine, and control circuitry specifically associated with the pump.	• failure to start on demand • failure to run • gross leakage
Motor Operated Valves	Includes valve, motor operator and the control circuitry, and electric power components specifically associated wtih the valve.(1)	• failure to open on demand • failure to remain open
Control Valves (Air Operated)	Includes the valve, the air actuator, and the control circuitry specifically associated with the valve.	• Failure to go to the "fail safe" position on signal • failure to provide control capability
Check Valve	Includes the check valve only	• failure to open
Relief Valve	Includes the relief valve only	• stuck open
Circuit Breaker/ Contactor (RPS)	The circuit breakers that provide power to the control rod drive mechanisms.	• failure to open
Relay (RPS)	The relays that actuate the trip breakers on signal from trip module.	• failure to open
Trip Module/	Includes the sensors, cables, bistables, and relays that measure plant parameters such as reactor coolant pressure and send a trip signal to trip breakers.	• failure to send trip signal when plant parameters require

Table 2. (Continued)

Component	Boundary	Failure Modes of Concern
Actuation Channel/ Subchannel	Includes the sensors, cables, bistables, and relays that measure plant parameters and send out an Engineered Safety Feature actuation signal.	• failure to send ESAS signal when required
Battery	Includes the battery and the battery charger.	• failure to provide DC power to components requiring DC power (given loss of AC power)
Diesel Generator	Includes the diesel and its support sytems (lube oil cooling, fuel supply, etc.).	• failure to provide AC power to components requiring AC power (given loss of off-site power)
Room Coolers	Includes the fan and cooling coils that provide room cooling to pump rooms.	• failure to cool pump room

(1) The electrical components specifically associated with the pump or motor operated valve would include the connector, cable, and circuit breaker that power the motor, but does not include the electric power distribution system that feeds the circuit breaker.

3.2.1 Oconee

highlight most imp systems to risk and the components within the systems

Table 3 shows the results for the Oconee PRA components. This table lists the individual components in order of importance as ranked by their risk aging sensitivity measure. As can be seen in Table 3 most of the components with the highest importance values are in the reactor protection system, the low pressure service water system, and the low pressure injection system. A number of the important components are electrical components including actuation channels, trip modules, circuit breakers, and contactors. The individual components are also grouped by type and system, and ranked by their aging sensitivity measure in Appendix A.

3.2.2 Calvert Cliffs

Table 4 shows the results for the Calvert Cliffs PRA. At this plant, the components with the highest aging sensitivity measures are components of the auxiliary feedwater system and the reactor protection system. Again, the components have been grouped by type and system, and these results are presented in Appendix A.

3.2.3 Grand Gulf

Table 5 shows the results for the Grand Gulf PRA. The components with the highest aging sensitivity measures are components of the service water system and the residual heat removal system. The components are grouped by type and system in Appendix A.

3.3 Combined Results

This section combines the results of the aging sensitivity measure calculations for individual components to provide an overall ranking. Two levels of ranking are provided.

Two ways of ranking

In the first ranking, components of the same type that are in the same system are grouped together, i.e., motor operated valves of the auxiliary feedwater system comprise one group. The aging sensitivity measure for the group is the sum of the aging sensitivity measures of the components in the group. The combined results provide an indication of which component groups have the greatest potential risk impact. This ranking of the component groups takes into account the importance of the individual components and the number of that type of component in each system.

The second ranking combines components of the same type but does not differentiate between systems. The aging sensitivity measure provided for the component type is the sum of the aging sensitivity measures of all the components of that type. The ranking is then a measure of the importance of a component type that takes into account the importance of individual components and the number of components of that type.

Table 3. Plant name: Oconee - Reactor type: PWR.

RANK	COMPONENT NAME	COMPONENT DESCRIPTION	RISK IMPACT OF COMPONENT UNAVAILABILITY (RANK)	COMPONENT TYPE S=PERIODICALLY TESTED O=CONTINUOUSLY MONITORED	RATE OF CHANGE OF COMPONENT UNAVAILABILITY WITH FAILURE RATE	RISK IMPACT OF COMPONENT AGING
1	CH 4	LPIS & STANDBY LPRS ACTUATION CHANNEL	1.0000E-03(4)	S	4.3379E-02	4.3379E-05
2	LPSW-P3A	STANDBY LPSW PUMP	7.6000E-04(5)	S	4.3379E-02	3.2968E-05
3	VP 2	STANDBY LPSW VACUUM PUMP	7.6000E-04(6)	S	4.3379E-02	3.2968E-05
4	CB A	RPS CIRCUIT BREAKER "A"	6.0000E-04(7)	S	4.3379E-02	2.6027E-05
5	CB B	RPS CIRCUIT BREAKER "B"	6.0000E-04(8)	S	4.3379E-02	2.6027E-05
6	LPSW-P3B	OPERATING LPSW PUMP	1.0000E-02(1)	O	2.2831E-03	2.2831E-05
7	VP1	OPERATING LPSW VACUUM PUMP	1.0000E-02(2)	O	2.2831E-03	2.2831E-05
8	Q	SAFETY RELIEF VALVE	3.5000E-04(9)	S	4.3379E-02	1.5183E-05
9	CB C	RPS CIRCUIT BREAKERS "C"	3.0000E-04(10)	S	4.3379E-02	1.3014E-05
10	CB D	RPS CIRCUIT BREAKERS "D"	3.0000E-04(11)	S	4.3379E-02	1.3014E-05
11	RTM 1	RPS REMOTE TRIP MODULE 1	3.0000E-04(12)	S	4.3379E-02	1.3014E-05
12	RTM 2	RPS REMOTE TRIP MODULE 2	3.0000E-04(13)	S	4.3379E-02	1.3014E-05
13	RTM 3	RPS REMOTE TRIP MODULE 3	3.0000E-04(14)	S	4.3379E-02	1.3014E-05
14	RTM 4	RPS REMOTE TRIP MODULE 4	3.0000E-04(15)	S	4.3379E-02	1.3014E-05
15	RPS E	RPS CONTACTOR "E"	3.0000E-04(16)	S	4.3379E-02	1.3014E-05
16	RPS F	RPS CONTACTOR "F"	3.0000E-04(17)	S	4.3379E-02	1.3014E-05
17	LP 17	LPIS A RECCR A MOTOR OPERATED VALVE	2.3000E-04(18)	S	4.3379E-02	9.9772E-06
18	LP P1A	LPIS A & ECCR A PUMP	2.3000E-04(19)	S	4.3379E-02	9.9772E-06
19	LP 18	LPIS B & ECCR B MOTOR OPERATED VALVE	2.3000E-04(20)	S	4.3379E-02	9.9772E-06
20	LP P1B	LPIS B & ECCR B PUMP	2.3000E-04(21)	S	4.3379E-02	9.9772E-06
21	CH 3	LPIS ACTUATION CHANNEL	2.3000E-04(22)	S	4.3379E-02	9.9772E-06
22	CF 12	LPIS A & ECCR A CHECK VALVE	2.3000E-04(23)	S	4.3379E-02	9.9772E-06
23	CF 14	LPIS B RECCR B CHECK VALVE	2.3000E-04(24)	S	4.3379E-02	9.9772E-06
24	LP 31	LPIS A RECCR A CHECK VALVE	2.3000E-04(25)	S	4.3379E-02	9.9772E-06
25	LP 12	LPIS A & ECCR A MOTOR OPERATED VALVE	2.3000E-04(27)	S	4.3379E-02	9.9772E-06
26	LP 48	LPIS A & ECCR A CHECK VALVE	2.3000E-04(29)	S	4.3379E-02	9.9772E-06
27	TEST A	LPIS A & ECCR A TEST VALVE	2.3000E-04(30)	S	4.3379E-02	9.9772E-06
28	LP 5	LPIS A& ECCR A MOTOR OPERATED VALVE	2.3000E-04(31)	S	4.3379E-02	9.9772E-06
29	LP 33	LPIS B & ECCR B CHECK VALVE	2.3000E-04(32)	S	4.3379E-02	9.9772E-06
30	LP 14	LPIS B & ECCR B MOTOR OPERATED VALVE	2.3000E-04(34)	S	4.3379E-02	9.9772E-06
31	LP 47	LPIS B & ECCR B CHECK VALVE	2.3000E-04(36)	S	4.3379E-02	9.9772E-06
32	TEST B	LPIS B & ECCR B TEST VALVE	2.3000E-04(37)	S	4.3379E-02	9.9772E-06
33	LP 8	LPIS B & ECCR B MOTOT OPERATED VALVE	2.3000E-04(38)	S	4.3379E-02	9.9772E-06
34	LP 22	LPIS B & ECCR B MOTOR OPERATED VALVE	2.2000E-04(39)	S	4.3379E-02	9.5434E-06
35	LP 30	LPIS B & ECCR CHECK VALVE	2.2000E-04(40)	S	4.3379E-02	9.5434E-06
36	LP 21	LPIS A & ECCR A MOTOR OPERATED VALVE	2.2000E-04(41)	S	4.3379E-02	9.5434E-06
37	LP 29	LPIS A & ECCR A CHECK VALVE	2.2000E-04(42)	S	4.3379E-02	9.5434E-06
38	CH 1	HPIS ACTUATION TRAIN	1.4000E-04(43)	S	4.3379E-02	6.0731E-06
39	HP 24	HPIS A MOTOR OPERATED VALVE	1.4000E-04(44)	S	4.3379E-02	6.0731E-06
40	HP 101	HPIS A CHECK VALVE	1.4000E-04(45)	S	4.3379E-02	6.0731E-06
41	HP 26	HPIS A MOTOR OPERATED VALVE	1.4000E-04(46)	S	4.3379E-02	6.0731E-06
42	LP 19	ECCR B SUMP VALVE	1.4000E-04(49)	S	4.3379E-02	6.0731E-06
43	LP 20	ECCR A SUMP VALVE	1.4000E-04(50)	S	4.3379E-02	6.0731E-06
44	FDW 232	AFWS CHECK VALVE	1.3000E-04(51)	S	4.3379E-02	5.6393E-06
45	FDW 317	AFWS CHECK VALVE	1.3000E-04(52)	S	4.3379E-02	5.6393E-06
46	FDW 315	AFWS AIR OPERATED VALVE	1.3000E-04(53)	S	4.3379E-02	5.6393E-06
47	FDW 233	AFWS CHECK VALVE	1.3000E-04(54)	S	4.3379E-02	5.6393E-06
48	FDW 314	AFWS CHECK VALVE	1.3000E-04(55)	S	4.3379E-02	5.6393E-06
49	FDW 316	AFWS AIR OPERATED VALVE	1.3000E-04(56)	S	4.3379E-02	5.6393E-06
50	HP 113	HPIS C CHECK VALVE	8.9000E-05(61)	S	4.3379E-02	3.8607E-06

Table 3. contd.

RANK	COMPONENT NAME	COMPONENT DESCRIPTION	RISK IMPACT OF COMPONENT UNAVAILABILITY (RANK)	COMPONENT TYPE S=PERIODICALLY TESTED O=CONTINUOUSLY MONITORED	RATE OF CHANGE OF COMPONENT UNAVAILABILITY WITH FAILURE RATE	RISK IMPACT OF COMPONENT AGING
51	HP 25	HPIS C MOTOR OPERATED VALVE	8.9000E-05(62)	S	4.3379E-02	3.8607E-06
52	HP 102	HPIS C CHECK VALVE	8.9000E-05(64)	S	4.3379E-02	3.8607E-06
53	HP P1C	HPIS C PUMP	8.9000E-05(65)	S	4.3379E-02	3.8607E-06
54	CH 2	STANDBY HPIS SUBSYSTEM ACTUATION CHANNEL	8.9000E-05(57)	S	4.3379E-02	3.8607E-06
55	HP 27	HPIS C MOTOR OPERATED VALVE	8.9000E-05(58)	S	4.3379E-02	3.8607E-06
56	BAT A	EPS BATTERY "A"	8.6000E-05(66)	S	4.3379E-02	3.7306E-06
57	BAT B	EPS BATTERY "B"	8.6000E-05(67)	S	4.3379E-02	3.7306E-06
58	EFP A	AFWS A ELECTRIC PUMP	7.0000E-05(68)	S	4.3379E-02	3.0365E-06
59	FDW 373	AFWS A CHECK VALVE	7.0000E-05(70)	S	4.3379E-02	3.0365E-06
60	FDW 370	AFWS A CHECK VALVE	7.0000E-05(71)	S	4.3379E-02	3.0365E-06
61	FDW 372	AFWS A MOTOR OPERATED VALVE	7.0000E-05(72)	S	4.3379E-02	3.0365E-06
62	EFP B	AFWS B ELECTRIC PUMP	7.0000E-05(73)	S	4.3379E-02	3.0365E-06
63	FDW 383	AFWS B CHECK VALVE	7.0000E-05(75)	S	4.3379E-02	3.0365E-06
64	FDW 380	AFWS B CHECK VALVE	7.0000E-05(76)	S	4.3379E-02	3.0365E-06
65	FDW 382	AFWS B MOTOR OPERATED VALVE	7.0000E-05(77)	S	4.3379E-02	3.0365E-06
66	TG 1	TURBOGENERATOR 1	3.6000E-05(78)	S	4.3379E-02	1.5616E-06
67	TG 2	TURBOGENERATOR 2	3.6000E-05(79)	S	4.3379E-02	1.5616E-06
68	HP 1AB	HPIS A OPERATING PUMP(S)	1.4000E-04(48)	O	2.2831E-03	3.1963E-07
69	MS 93	AFWS T TURBINE AIR OPERATED VALVE	2.0000E-06(82)	S	4.3379E-02	8.6758E-08
70	MS 94	AFWS T TURBINE OVERSPEED VALVE	2.0000E-06(83)	S	4.3379E-02	8.6758E-08
71	MS 95	AFWS T TURBINE GOVERNOR VALVE	2.0000E-06(84)	S	4.3379E-02	8.6758E-08
72	MS 87	AFWS T TURBINE AIR OPERATED VALVE	2.0000E-06(85)	S	4.3379E-02	8.6758E-08
73	EFP-TD	AFWS T TURBINE PUMP	2.0000E-06(86)	S	4.3379E-02	8.6758E-08
74	C 156	AFWS T MOTOR OPERATED VALVE	2.0000E-06(89)	S	4.3379E-02	8.6758E-08
75	LPSW 137	AFWS T & LPSW MOTOR OPERATED VALVE	2.0000E-06(90)	S	4.3379E-02	8.6758E-08
76	HP 111	HPIS C MANUAL VALVE	8.9000E-05(63)	S	0.0000E+00	0.0000E+00
77	HP 114	HPIS C MANUAL VALVE	8.9000E-05(60)	S	0.0000E+00	0.0000E+00
78	HP 148	HPIS C MANUAL VALVE	8.9000E-05(59)	S	0.0000E+00	0.0000E+00
79	LP 16	LPIS B & ECCR B MANUAL VALVE	2.3000E-04(35)	S	0.0000E+00	0.0000E+00
80	LP 13	LPIS B & ECCR B MANUAL VALVE	2.3000E-04(33)	S	0.0000E+00	0.0000E+00
81	LP 15	LPIS A & ECCR A MANUAL VALVE	2.3000E-04(28)	S	0.0000E+00	0.0000E+00
82	LP 11	LPIS A & ECCR A MANUAL VALVE	2.3000E-04(26)	S	0.0000E+00	0.0000E+00
83	LP 28	MANUAL VALVE FOR LPIS & HPIS	3.6000E-03(3)	S	0.0000E+00	0.0000E+00
84	C 157	AFWS T MANUAL VALVE	2.0000E-06(88)	S	0.0000E+00	0.0000E+00
85	FDW 88	AFWS T MANUAL VALVE	2.0000E-06(87)	S	0.0000E+00	0.0000E+00
86	MS 91	AFWS T MANUAL VALVE	2.0000E-06(81)	S	0.0000E+00	0.0000E+00
87	MS 90	AFWS T MANUAL VALVE	2.0000E-06(80)	S	0.0000E+00	0.0000E+00
88	C 576	AFWS B MANUAL VALVE	7.0000E-05(74)	S	0.0000E+00	0.0000E+00
89	C 575	AFWS A MANUAL VALVE	7.0000E-05(69)	S	0.0000E+00	0.0000E+00
90	HP 118	HPIS A MANUAL VALVE	1.4000E-04(47)	S	0.0000E+00	0.0000E+00

Table 4. Plant name: Colvert Cliffs - Reactor type: PWR

RANK	COMPONENT NAME	COMPONENT DESCRIPTION	RISK IMPACT OF COMPONENT UNAVAILABILITY (RANK)	COMPONENT TYPE S=PERIODICALLY TESTED O=CONTINUOUSLY MONITORED	RATE OF CHANGE OF COMPONENT UNAVAILABILITY WITH FAILURE RATE	RISK IMPACT OF COMPONENT AGING.
1	TP21	AFWS TURBINE PUMP	1.6000E-02(3)	S	4.3379E-02	6.9406E-04
2	TP22	AFWS TURBINE PUMP	1.6000E-02(4)	S	4.3379E-02	6.9406E-04
3	CV4511	AFWS CONTROL VALVE	1.6000E-02(5)	S	4.3379E-02	6.9406E-04
4	CV4512	AFWS CONTROL VALVE	1.6000E-02(6)	S	4.3379E-02	6.9406E-04
5	MOV4071	AFWS MOTOR OPERATED VALVE	1.6000E-02(7)	S	4.3379E-02	6.9406E-04
6	MOV4070	AFWS MOTOR OPERATED VALVE	1.6000E-02(8)	S	4.3379E-02	6.9406E-04
7	P3	AFWS CHECK VALVE	1.6000E-02(12)	S	4.3379E-02	6.9406E-04
8	S5	AFWS CHECK VALVE	1.6000E-02(13)	S	4.3379E-02	6.9406E-04
9	P5	AFWS CHECK VALVE	1.6000E-02(17)	S	4.3379E-02	6.9406E-04
10	S7	AFWS CHECK VALVE	1.6000E-02(18)	S	4.3379E-02	6.9406E-04
11	H5	AFWS CHECK VALVE	1.6000E-02(20)	S	4.3379E-02	6.9406E-04
12	H6	AFWS CHECK VALVE	1.6000E-02(22)	S	4.3379E-02	6.9406E-04
13	S3	AFWS CHECK VALVE	1.6000E-02(23)	S	4.3379E-02	6.9406E-04
14	S4	AFWS CHECK VALVE	1.6000E-02(24)	S	4.3379E-02	6.9406E-04
15	K1	RPS RELAY	1.2000E-02(25)	S	4.3379E-02	5.2055E-04
16	K2	RPS RELAY	1.2000E-02(26)	S	4.3379E-02	5.2055E-04
17	K3	RPS RELAY	1.2000E-02(27)	S	4.3379E-02	5.2055E-04
18	K4	RPS RELAY	1.2000E-02(28)	S	4.3379E-02	5.2055E-04
19	1A	RPS CIRCUIT BREAKER	9.0000E-03(29)	S	4.3379E-02	3.9041E-04
20	2A	RPS CIRCUIT BREAKER	9.0000E-03(30)	S	4.3379E-02	3.9041E-04
21	3A	RPS CIRCUIT BREAKER	9.0000E-03(31)	S	4.3379E-02	3.9041E-04
22	4A	RPS CIRCUIT BREAKER	9.0000E-03(32)	S	4.3379E-02	3.9041E-04
23	1B	RPS CIRCUIT BREAKER	9.0000E-03(33)	S	4.3379E-02	3.9041E-04
24	2B	RPS CIRCUIT BREAKER	9.0000E-03(34)	S	4.3379E-02	3.9041E-04
25	3B	RPS CIRCUIT BREAKER	9.0000E-03(35)	S	4.3379E-02	3.9041E-04
26	4B	RPS CIRCUIT BREAKER	9.0000E-03(36)	S	4.3379E-02	3.9041E-04
27	MOV659	HPIS #21 & #23 MOTOR OPERATED VALVE	4.4000E-03(38)	S	4.3379E-02	1.9087E-04
28	MOV660	HPIS #21 & #23 MOTOR OPERATED VALVE	4.4000E-03(39)	S	4.3379E-02	1.9087E-04
29	D12ST	EPS DIESEL GENERATOR #12	2.2000E-03(40)	S	4.3379E-02	9.5434E-05
30	STB7	STAS SUBCHANNEL B7	1.9000E-03(41)	S	4.3379E-02	8.2420E-05
31	S22	SALT #22 PUMP	1.5000E-03(42)	S	4.3379E-02	6.5068E-05
32	D21ST	EPS DIESEL GENERATOR #21	1.5000E-03(43)	S	4.3379E-02	6.5068E-05
33	BAT21	EPS BATTERY #21	1.5000E-03(44)	S	4.3379E-02	6.5068E-05
34	CV5152	SALT #22 CONTROL VALVE	1.5000E-03(45)	S	4.3379E-02	6.5068E-05
35	CV5153	SALT #22 CONTROL VALVE	1.5000E-03(46)	S	4.3379E-02	6.5068E-05
36	CV5212	SALT #22 CONTROL VALVE	1.5000E-03(47)	S	4.3379E-02	6.5068E-05
37	S#22	SWS #22 PUMP	1.5000E-03(48)	S	4.3379E-02	6.5068E-05
38	CV5162	SALT #22 CONTROL VALVE	6.6000E-04(49)	S	4.3379E-02	2.8630E-05
39	CV5208	SALT #22 CONTROL VALVE	6.6000E-04(50)	S	4.3379E-02	2.8630E-05
40	MOV4143	HPIS #21 & LPIS #21 & HPRS #21 MOTOR OPERATED VALV	6.6000E-04(51)	S	4.3379E-02	2.8630E-05
41	C55	HPIS #21 & LPIS #21 & HPRS #21 CHECK VALVE	6.6000E-04(52)	S	4.3379E-02	2.8630E-05
42	C37	HPIS #21 & HPRS #21 CHECK VALVE	6.6000E-04(59)	S	4.3379E-02	2.8630E-05
43	C64	HPIS #21 & HPRS #21 CHECK VALVE	6.6000E-04(60)	S	4.3379E-02	2.8630E-05
44	HP21	HPIS #21 & HPRS #21 PUMP	6.6000E-04(61)	S	4.3379E-02	2.8630E-05
45	STA2	STAS SUBCHANNEL A2	6.6000E-04(62)	S	4.3379E-02	2.8630E-05
46	MOV4142	HPIS #23 & LPIS #22 & HPRS #22 MOTOR OPERATED VALV	4.7000E-04(63)	S	4.3379E-02	2.0388E-05
47	C66	HPIS #23 & LPIS #22 & HPRS #22 CHECK VALVE	4.7000E-04(64)	S	4.3379E-02	2.0388E-05
48	STB2	STAS SUBCHANNEL B2	4.6000E-04(66)	S	4.3379E-02	1.9954E-05
49	C61	HPIS #23 & HPRS #22 CHECK VALVE	4.5000E-04(70)	S	4.3379E-02	1.9521E-05
50	C39	HPIS #23 & HPRS #22 CHECK VALVE	4.5000E-04(69)	S	4.3379E-02	1.9521E-05

Table 4. contd.

RANK	COMPONENT NAME	COMPONENT DESCRIPTION	RISK IMPACT OF COMPONENT UNAVAILABILITY (RANK)	COMPONENT TYPE S=PERIODICALLY TESTED O=CONTINUOUSLY MONITORED	RATE OF CHANGE OF COMPONENT UNAVAILABILITY WITH FAILURE RATE	RISK IMPACT OF COMPONENT AGING
51	HP23	HPIS #23 & HPRS #22 PUMP	4.5000E-04(71)	S	4.3379E-02	1.9521E-05
52	R21	SALT #21 ROOM COOLER	4.4000E-04(72)	S	4.3379E-02	1.9087E-05
53	C21	ECCR #21 CHECK VALVE	4.4000E-04(73)	S	4.3379E-02	1.9087E-05
54	MOV4144	ECCR #21 MOTOR OPERATED VALVE	4.4000E-04(74)	S	4.3379E-02	1.9087E-05
55	MOV4145	ECCR #22 MOTOR OPERATED VALVE	3.3000E-04(77)	S	4.3379E-02	1.4315E-05
56	C20	ECCR #22 CHECK VALVE	3.3000E-04(78)	S	4.3379E-02	1.4315E-05
57	R22	SALT #22 ROOM COOLER	3.3000E-04(79)	S	4.3379E-02	1.4315E-05
58	CC22	CCW STANBY PUMP	3.1000E-04(80)	S	4.3379E-02	1.3447E-05
59	C115	CCW CHECK VALVE	3.1000E-04(83)	S	4.3379E-02	1.3447E-05
60	STA1	STAS SUBCHANNEL A1	2.8000E-04(84)	S	4.3379E-02	1.2146E-05
61	MOV656	HPIS #21 MOTOR OPERATED VALVE	2.7000E-04(85)	S	4.3379E-02	1.1712E-05
62	CV5160	SALT #21 CONTROL VALVE	2.5000E-04(86)	S	4.3379E-02	1.0845E-05
63	CV5206	SALT #21 CONTROL VALVE	2.5000E-04(87)	S	4.3379E-02	1.0845E-05
64	CV3824	CCW CONTROL VALVE	2.5000E-04(88)	S	4.3379E-02	1.0845E-05
65	MOV654	HPIS #23 MOTOR OPERATED VALVE	2.1000E-04(91)	S	4.3379E-02	9.1096E-06
66	STB1	STAS SUBCHANNEL B1	2.1000E-04(92)	S	4.3379E-02	9.1096E-06
67	S21	SALT #21 PUMP	1.8000E-04(93)	S	4.3379E-02	7.8082E-06
68	RASA1	RAS SUBCHANNEL A1	1.5000E-04(94)	S	4.3379E-02	6.5068E-06
69	CV657	LPIS CONTROL VALVE	1.1000E-04(95)	S	4.3379E-02	4.7717E-06
70	MOV658	LPIS CONTROL VALVE	1.1000E-04(96)	S	4.3379E-02	4.7717E-06
71	CV306	LPIS CONTROL VALVE	1.1000E-04(97)	S	4.3379E-02	4.7717E-06
72	RASB1	RAS SUBCHANNEL B1	1.0000E-04(98)	S	4.3379E-02	4.3379E-06
73	CV5210	SALT #21 CONTROL VALVE	7.0000E-05(99)	S	4.3379E-02	3.0365E-06
74	CV5150	SALT #21 CONTROL VALVE	7.0000E-05(100)	S	4.3379E-02	3.0365E-06
75	SW21	SW #21 PUMP	7.0000E-05(101)	S	4.3379E-02	3.0365E-06
76	CC21	CCW OPERATING PUMP	4.1000E-04(75)	O	2.2831E-03	9.3607E-07
77	STA3	STAS SUBCHANNEL A3	2.4000E-06(102)	S	4.3379E-02	1.0411E-07
78	STB3	STAS SUBCHANNEL B3	2.4000E-06(103)	S	4.3379E-02	1.0411E-07
79	C41	LPIS #22 & LPRS #22 CHECK VALVE	2.2000E-06(107)	S	4.3379E-02	9.5434E-08
80	C63	LPIS #22 & LPRS #22 CHECK VALVE	2.2000E-06(108)	S	4.3379E-02	9.5434E-08
81	LP22	LPIS #22 & LPRS #22 PUMP	2.2000E-06(109)	S	4.3379E-02	9.5434E-08
82	BAT12	EPS BATTERY #12	2.2000E-06(110)	S	4.3379E-02	9.5434E-08
83	BAT22	EPS BATTERY #22	2.2000E-06(111)	S	4.3379E-02	9.5434E-08
84	C35	LPIS #21 & LPRS #21 CHECK VALVE	2.0000E-06(115)	S	4.3379E-02	8.6758E-08
85	C56	LPIS #21 & LPRS #21 CHECK VALVE	2.0000E-06(116)	S	4.3379E-02	8.6758E-08
86	LP21	LPIS #21 & LPRS #21 PUMP	2.0000E-06(117)	S	4.3379E-02	8.6758E-08
87	M108	CCW MANUAL VALVE	6.6000E-04(56)	S	0.0000E+00	0.0000E+00
88	M107	CCW MANUAL VALVE	6.6000E-04(55)	S	0.0000E+00	0.0000E+00
89	M106	CCW MANUAL VALVE	6.6000E-04(54)	S	0.0000E+00	0.0000E+00
90	M105	CCW MANUAL VALVE	6.6000E-04(53)	S	0.0000E+00	0.0000E+00
91	M111	CCW MANUAL VALVE	5.3000E-03(37)	S	0.0000E+00	0.0000E+00
92	M2	AFWS MANUAL VALVE	1.6000E-02(21)	S	0.0000E+00	0.0000E+00
93	M1	AFWS MANUAL VALVE	1.6000E-02(19)	S	0.0000E+00	0.0000E+00
94	S8	AFWS MANUAL VALVE	1.6000E-02(16)	S	0.0000E+00	0.0000E+00
95	P6	AFWS MANUAL VALVE	1.6000E-02(15)	S	0.0000E+00	0.0000E+00
96	P2	AFWS MANUAL VALVE	1.6000E-02(14)	S	0.0000E+00	0.0000E+00
97	S6	AFWS MANUAL VALVE	1.6000E-02(11)	S	0.0000E+00	0.0000E+00
98	P4	AFWS MANUAL VALVE	1.6000E-02(10)	S	0.0000E+00	0.0000E+00
99	P1	AFWS MANUAL VALVE	1.6000E-02(9)	S	0.0000E+00	0.0000E+00
100	C4	AFWS MANUAL VALVE	5.4000E-01(2)	S	0.0000E+00	0.0000E+00

Table 5. Plant name: Grand Gulf - Reactor type: BWR

RANK	COMPONENT NAME	COMPONENT DESCRIPTION	RISK IMPACT OF COMPONENT UNAVAILABILITY (RANK)	COMPONENT TYPE S=PERIODICALLY TESTED O=CONTINUOUSLY MONITTORED	RATE OF CHANGE OF COMPONENT UNAVAILABILITY WITH FAILURE RATE	RISK IMPACT OF COMPONENT AGING
1	F001AA	SSWS A & RHR A & SPMS A MOTOR OPERATED VALVE	7.3000E-04(1)	S	4.3379E-02	3.1667E-05
2	F001BB	SSWS B & RHR B & SPMS B MOTOR OPERATED VAVLE	7.3000E-04(2)	S	4.3379E-02	3.1667E-05
3	C001AA	SSWS A PUMP	6.7000E-04(3)	S	4.3379E-02	2.9064E-05
4	C001BB	SSWS B PUMP	6.7000E-04(4)	S	4.3379E-02	2.9064E-05
5	F005AA	SSWS A MOTOR OPERATED VALVE	6.7000E-04(5)	S	4.3379E-02	2.9064E-05
6	F005BB	SSWS B MOTOR OPERATED VALVE	6.7000E-04(7)	S	4.3379E-02	2.9064E-05
7	SAC	SSWS A ACTUATION AND CONTROL CIRCUIT	6.7000E-04(8)	S	4.3379E-02	2.9064E-05
8	SBC	SSWS B ACTUATION AND CONTROL CIRCUIT	6.7000E-04(10)	S	4.3379E-02	2.9064E-05
9	F008A	SSWS A CHECK VALVE	6.7000E-04(11)	S	4.3379E-02	2.9064E-05
10	F008B	SSWS B CHECK VALVE	6.7000E-04(12)	S	4.3379E-02	2.9064E-05
11	F014AA	RHR A MOTOR OPERATED VALVE	5.8000E-04(13)	S	4.3379E-02	2.5160E-05
12	F068AA	RHR A MOTOR OPERATED VALVE	5.8000E-04(14)	S	4.3379E-02	2.5160E-05
13	F014BB	RHR B MOTOR OPERATED VALVE	5.8000E-04(15)	S	4.3379E-02	2.5160E-05
14	F068BB	RHR B MOTOR OPERATED VALVE	5.8000E-04(16)	S	4.3379E-02	2.5160E-05
15	BATA	EPS BATTEERY A	4.5000E-04(21)	S	4.3379E-02	1.9521E-05
16	LRACT	LPCS & LPCIS A & RHR A INITIATING LOGIC CIRCUIT	3.3000E-04(22)	S	4.3379E-02	1.4315E-05
17	HCACT	LPCIS C & LPCIS B & RHR B INITIATING LOGIC CIRCUIT	3.0000E-04(23)	S	4.3379E-02	1.3014E-05
18	F003AA	RHR A MOTOR OPERATED VALVE	2.8000E-04(24)	S	4.3379E-02	1.2146E-05
19	F047AA	RHR A MOTOR OPERATED VALVE	2.8000E-04(25)	S	4.3379E-02	1.2146E-05
20	F003BB	RHR B MOTOR OPERATED VALVE	2.8000E-04(26)	S	4.3379E-02	1.2146E-05
21	F047BB	RHR B MOTOR OPERATED VALVE	2.8000E-04(27)	S	4.3379E-02	1.2146E-05
22	C002BB	LPCIS B & RHR B PUMP	2.8000E-04(28)	S	4.3379E-02	1.2146E-05
23	F004BB	LPCIS B & RHR B MOTOR OPERATED VALVE	2.8000E-04(29)	S	4.3379E-02	1.2146E-05
24	F031B	LPCIS B & RHR B CHECK VALVE	2.8000E-04(31)	S	4.3379E-02	1.2146E-05
25	C002AA	LPCIS A & RHR A PUMP	2.6000E-04(32)	S	4.3379E-02	1.1279E-05
26	F024AA	RHR A MOTOR OPERATED VALVE	2.6000E-04(33)	S	4.3379E-02	1.1279E-05
27	F024BB	RHR B MOTOR OPERATED VALVE	2.6000E-04(34)	S	4.3379E-02	1.1279E-05
28	F004AA	LPCIS A & RHR A MOTOR OPERATED VALVE	2.6000E-04(35)	S	4.3379E-02	1.1279E-05
29	F048AA	RHR A MOTOR OPERATED VALVE	2.6000E-04(36)	S	4.3379E-02	1.1279E-05
30	F048BB	RHR B MOTOR OPERATED VALVE	2.6000E-04(37)	S	4.3379E-02	1.1279E-05
31	F031A	LPCIS A & RHR A CHECK VALVE	2.6000E-04(46)	S	4.3379E-02	1.1279E-05
32	C001	RCISC PUMP	1.0000E-04(48)	S	4.3379E-02	4.3379E-06
33	F013A	RCICS MOTOR OPERATED VALVE	1.0000E-04(49)	S	4.3379E-02	4.3379E-06
34	F045A	RCICS MOTOR OPERATED VALVE	1.0000E-04(50)	S	4.3379E-02	4.3379E-06
35	F068A	RCICS MOTOR OPERATED VALVE	1.0000E-04(51)	S	4.3379E-02	4.3379E-06
36	F010A	RCICS MOTOR OPERATED VALVE	1.0000E-04(52)	S	4.3379E-02	4.3379E-06
37	F064A	RCICS MOTOR OPERATED VALVE	1.0000E-04(53)	S	4.3379E-02	4.3379E-06
38	F063B	RCICS MOTOR OPERATED VALVE	1.0000E-04(54)	S	4.3379E-02	4.3379E-06
39	TTV	RCICS TRIP THROTTLE VALVE	1.0000E-04(55)	S	4.3379E-02	4.3379E-06
40	TGV	RCICS TURBINE GOVERNING VALVE	1.0000E-04(56)	S	4.3379E-02	4.3379E-06
41	C002	RCICS TURBINE	1.0000E-04(57)	S	4.3379E-02	4.3379E-06
42	RACT	RCICS ACTUATING CIRCUIT	1.0000E-04(58)	S	4.3379E-02	4.3379E-06
43	F040	RCICS CHECK VALVE	1.0000E-04(59)	S	4.3379E-02	4.3379E-06
44	F066	RCICS CHECK VALVE	1.0000E-04(60)	S	4.3379E-02	4.3379E-06
45	F065	RCICS CHECK VALVE	1.0000E-04(61)	S	4.3379E-02	4.3379E-06
46	F208	RCICS CHECK VALVE	1.0000E-04(62)	S	4.3379E-02	4.3379E-06
47	HACT	HPCS ACTUATING CIRCUIT	6.5000E-05(80)	S	4.3379E-02	2.8196E-06
48	F001C	HPCS MOTOR OPERATED VALVE	6.5000E-05(79)	S	4.3379E-02	2.8196E-06
49	F002	HPCS CHECK VALVE	6.5000E-05(78)	S	4.3379E-02	2.8196E-06
50	C001C	HPCS PUMP	6.5000E-05(77)	S	4.3379E-02	2.8196E-06

Table 5. contd.

RANK	COMPONENT NAME	COMPONENT DESCRIPTION	RISK IMPACT OF COMPONENT UNAVAILABILITY (RANK)	COMPONENT TYPE S=PERIODICALLY TESTED O=CONTINUOUSLY MONITORED	RATE OF CHANGE OF COMPONENT UNAVAILABILITY WITH FAILURE RATE	RISK IMPACT OF COMPONENT AGING
51	F024	HPCS CHECK VALVE	6.5000E-05(76)	S	4.3379E-02	2.8196E-06
52	F004C	HPCS MOTOR OPERATED VALVE	6.5000E-05(75)	S	4.3379E-02	2.8196E-06
53	F005	HPCS CHECK VALVE	6.5000E-05(74)	S	4.3379E-02	2.8196E-06
54	P	SAFETY RELIEF VALVES	6.1000E-05(81)	S	4.3379E-02	2.6461E-06
55	BATB	EPS BATTERY B	9.4000E-05(72)	S	4.3379E-02	4.0776E-06
56	F002AA	SPWS A MOTOR OPERATED VALVE	4.4000E-05(82)	S	4.3379E-02	1.9087E-06
57	F018BB	SSWS B MOTOR OPERATED VALVE	9.4000E-05(70)	S	4.3379E-02	4.0776E-06
58	DIESEL2	EPS DIESEL GENERATOR #2	9.4000E-05(69)	S	4.3379E-02	4.0776E-06
59	F002BB	SPWS B MOTOR OPERATED VALVE	4.4000E-05(83)	S	4.3379E-02	1.9087E-06
60	F018AA	SSWS A MOTOR OPERATED VALVE	1.0000E-04(67)	S	4.3379E-02	4.3379E-06
61	DIESEL1	EPS DIESEL GENERATOR #1	1.0000E-04(66)	S	4.3379E-02	4.3379E-06
62	F011	RCICS CHECK VALVE	1.0000E-04(65)	S	4.3379E-02	4.3379E-06
63	SAACC	SPWS A ACTUATION & CONTROL CIRCUIT	4.4000E-05(84)	S	4.3379E-02	1.9087E-06
64	SBACC	SPWS B ACTUATION & CONTROL CIRCUIT	4.4000E-05(85)	S	4.3379E-02	1.9087E-06
65	DIESEL3	EPS DIESEL GENERATOR #3	2.7000E-05(86)	S	4.3379E-02	1.1712E-06
66	BATC	EPS BATTERY C	2.7000E-05(87)	S	4.3379E-02	1.1712E-06
67	C002C	SSWS C PUMP	2.7000E-05(92)	S	4.3379E-02	1.1712E-06
68	F012	SSWS C CHECK VALVE	2.7000E-05(93)	S	4.3379E-02	1.1712E-06
69	F011C	SSWS C MOTOR OPERATED VALVE	2.7000E-05(95)	S	4.3379E-02	1.1712E-06
70	SCC	SSWS C ACTUATION & CONTROL CIRCUIT	2.7000E-05(96)	S	4.3379E-02	1.1712E-06
71	F087AA	RHR A MOTOR OPERATED VALVE	1.9000E-05(97)	S	4.3379E-02	8.2420E-07
72	F052AA	RHR A MOTOR OPERATED VALVE	1.9000E-05(98)	S	4.3379E-02	8.2420E-07
73	F026AA	RHR A MOTOR OPERATED VALVE	1.9000E-05(99)	S	4.3379E-02	8.2420E-07
74	F054A	RHR A CHECK VALVE	1.9000E-05(100)	S	4.3379E-02	8.2420E-07
75	F087BB	RHR B MOTOR OPERATED VALVE	1.9000E-05(101)	S	4.3379E-02	8.2420E-07
76	F052BB	RHR B MOTOR OPERATED VALVE	1.9000E-05(102)	S	4.3379E-02	8.2420E-07
77	F026BB	RHR B MOTOR OPERATED VALVE	1.9000E-05(103)	S	4.3379E-02	8.2420E-07
78	F054B	RHR B CHECK VALVE	1.9000E-05(104)	S	4.3379E-02	8.2420E-07
79	F241	LPCIS C CHECK VALVE	1.6000E-05(106)	S	4.3379E-02	6.9406E-07
80	F242B	LPCIS C MOTOR OPERATED VALVE	1.6000E-05(107)	S	4.3379E-02	6.9406E-07
81	F031C	LPCIS C CHECK VALVE	1.6000E-05(109)	S	4.3379E-02	6.9406E-07
82	C002CB	LPCIS C PUMP	1.6000E-05(110)	S	4.3379E-02	6.9406E-07
83	F004CB	LPCIS C MOTOR OPERATED VALVE	1.6000E-05(111)	S	4.3379E-02	6.9406E-07
84	F041B	LPCIS B CHECK VALVE	1.6000E-05(113)	S	4.3379E-02	6.9406E-07
85	F042BB	LPCIS B MOTOR OPERATED VALVE	1.6000E-05(114)	S	4.3379E-02	6.9406E-07
86	F027BB	LPCIS B MOTOR OPERATED VALVE	1.6000E-05(115)	S	4.3379E-02	6.9406E-07
87	F001A	LPCS MOTOR OPERATED VALVE	1.4000E-05(116)	S	4.3379E-02	6.0731E-07
88	C001A	LPCS PUMP	1.4000E-05(117)	S	4.3379E-02	6.0731E-07
89	F003	LPCS CHECK VALVE	1.4000E-05(118)	S	4.3379E-02	6.0731E-07
90	F005A	LPCS MOTOR OPERATED VALVE	1.4000E-05(119)	S	4.3379E-02	6.0731E-07
91	F006	LPCS CHECK VALVE	1.4000E-05(120)	S	4.3379E-02	6.0731E-07
92	F041A	LPCIS A CHECK VALVE	1.5000E-06(123)	S	4.3379E-02	6.5068E-08
93	F042AA	LPCIS A MOTOR OPERATED VALVE	1.5000E-06(124)	S	4.3379E-02	6.5068E-08
94	F027AA	LPCIS A MOTOR OPERATED VALVE	1.5000E-06(125)	S	4.3379E-02	6.5068E-08
95	F200	RCICS MANUAL VALVE	1.0000E-04(63)	S	0.0000E+00	0.0000E+00
96	F029A	LPCIS A & RHR A MANUAL VALVE	2.6000E-04(47)	S	0.0000E+00	0.0000E+00
97	F083B	RHR B MANUAL VALVE	2.6000E-04(45)	S	0.0000E+00	0.0000E+00
98	F083A	RHR A MANUAL VALVE	2.6000E-04(44)	S	0.0000E+00	0.0000E+00
99	F210B	RHR B MANUAL VALVE	2.6000E-04(43)	S	0.0000E+00	0.0000E+00
100	F210A	RHR A MANUAL VALVE	2.6000E-04(42)	S	0.0000E+00	0.0000E+00

Table 5. contd.

RANK	COMPONENT NAME	COMPONENT DESCRIPTION	RISK IMPACT OF COMPONENT UNAVAILABILITY (RANK)	COMPONENT TYPE S=PERIODICALLY TESTED U=CONTINUOUSLY MONITORED	RATE OF CHANGE OF COMPONENT UNAVAILABILITY WITH FAILURE RATE	RISK IMPACT OF COMPONENT AGING
101	F103B	RHR B MANUAL VALVE	2.6000E-04(41)	S	0.0000E+00	0.0000E+00
102	F102B	RHR B MANUAL VALVE	2.6000E-04(40)	S	0.0000E+00	0.0000E+00
103	F103A	RHR A MANUAL VALVE	2.6000E-04(39)	S	0.0000E+00	0.0000E+00
104	F102A	RHR A MANUAL VALVE	2.6000E-04(38)	S	0.0000E+00	0.0000E+00
105	F029B	LPCIS B & RHR B MANUAL VALVE	2.8000E-04(30)	S	0.0000E+00	0.0000E+00
106	F130B	RHR B MANUAL VALVE	5.8000E-04(20)	S	0.0000E+00	0.0000E+00
107	F120B	RHR B MANUAL VALVE	5.8000E-04(19)	S	0.0000E+00	0.0000E+00
108	F130A	RHR A MANUAL VALVE	5.8000E-04(18)	S	0.0000E+00	0.0000E+00
109	F120A	RHR A MANUAL VALVE	5.8000E-04(17)	S	0.0000E+00	0.0000E+00
110	F199B	SSWS B MANUAL VALVE	6.7000E-04(9)	S	0.0000E+00	0.0000E+00
111	F199A	SSWS A MANUAL VALVE	6.7000E-04(6)	S	0.0000E+00	0.0000E+00
112	F039A	LPCIS A MANUAL VALVE	1.5000E-06(122)	S	0.0000E+00	0.0000E+00
113	F007	LPCS MANUAL VALVL	1.4000E-05(121)	S	0.0000E+00	0.0000E+00
114	F039B	LPCIS B MANUAL VALVE	1.6000E-05(112)	S	0.0000E+00	0.0000E+00
115	F029C	LPCIS C MANUAL VALVE	1.6000E-05(108)	S	0.0000E+00	0.0000E+00
116	F239	LPCIS C MANUAL VALVE	1.6000E-05(105)	S	0.0000E+00	0.0000E+00
117	F013	SSWS C MANUAL VALVE	2.7000E-05(94)	S	0.0000E+00	0.0000E+00
118	F186B	SSWS C MANUAL VALVE	2.7000E-05(91)	S	0.0000E+00	0.0000E+00
119	F186A	SSWS C MANUAL VALVE	2.7000E-05(90)	S	0.0000E+00	0.0000E+00
120	F185B	SSWS C MANUAL VALVE	2.7000E-05(89)	S	0.0000E+00	0.0000E+00
121	F185A	SSWS C MANUAL VALVE	2.7000E-05(88)	S	0.0000E+00	0.0000E+00
122	F205	HPCS MANUAL VALVE	6.5000E-05(73)	S	0.0000E+00	0.0000E+00
123	F023B	SSWS B MANUAL VALVE	9.4000E-05(71)	S	0.0000E+00	0.0000E+00
124	F023A	SSWS A MANUAL VALVE	1.0000E-04(68)	S	0.0000E+00	0.0000E+00
125	F016	RCICS MANUAL VALVE	1.0000E-04(64)	S	0.0000E+00	0.0000E+00

In the last part of this section, the results of the two PWR's are combined to give an overall ranking for PWR components.

3.3.1 Oconee

Table 6 shows the combined results of components of the same type and system at Oconee. The component groups are ranked from highest to lowest. The table shows that the component groups with the highest potential risk impact are service water pumps, low pressure emergency core cooling system motor operated valves and check valves, reactor protection system circuit breakers, and engineered safety feature actuation system actuators.

Table 7 shows the ranking for component types without differentiating between systems. The types of components with the most potential risk impact are pumps, check valves, actuation channels/trip modules, motor operated valves, and circuit breakers/contactors.

3.3.2 Calvert Cliffs

Table 8 shows the combined results for component groups at the Calvert Cliffs. The component groups with the highest potential risk significance are all in the auxiliary feedwater system (check valves, motor operated valves, and pumps) and the reactor protection system (circuit breakers and trip relays).

Table 9 shows the results of aging sensitivity measure calculations for component types. Check valves have the highest potential risk significance followed by circuit breakers, relays/actuation subchannels, motor operated valves, air operated control valves, and pumps.

3.3.3 Grand Gulf

Table 10 shows the combined results for component groups at the Grand Gulf. Motor operated valves of the low pressure emergency core cooling system and service water system and actuators of the engineered safety actuation system have the highest potential risk impacts as measured by the aging sensitivity measure.

Table 11 shows the ranking of the component types. Motor operated valves, check valves, actuators, and pumps have the highest values of the aging sensitivity measure.

Table 6. Aging sensitivity of component groups at Oconee.

Rank	Type	System	Aging Sensitivity (per reactor year)
1	Pump	Service Water	1.1×10^{-4}
2	Check Valve	Low Pressure ECC	9.8×10^{-5}
3	Circuit Breaker	Reactor Protection	7.8×10^{-5}
4	Motor Operated Valve	Low Pressure ECC	7.1×10^{-5}
5	Actuators	Safeguard Actuation	6.3×10^{-5}
6	Trip Modules	Reactor Protection	5.2×10^{-5}
7	Check Valves	Auxiliary Feedwater	3.3×10^{-5}
8	Contactor	Reactor Protection	2.6×10^{-5}
9	Pump	Low Pressure ECC	2.0×10^{-5}
10	Motor Operated Valve	High Pressure ECC	2.0×10^{-5}
11	Relief Valve	Reactor Pressure Control	1.5×10^{-5}
12	Control Valve (air operated)	Auxiliary Feedwater	1.2×10^{-5}
13	Batteries	Emergency Power	8.0×10^{-6}
14	Check Valves	High Pressure ECC	8.0×10^{-6}
15	Pump	Auxiliary Feedwater	6.1×10^{-6}
16	Motor Operated Valve	Auxiliary Feedwater	6.0×10^{-6}
17	Pump	High Pressure ECC	6.0×10^{-6}
18	Turbogenerator	Emergency Power	4.0×10^{-6}

Table 7. Aging sensitivity of component types at Oconee.

Rank	Component Type	Aging Sensitivity	% Contribution
1	Pumps	1.4×10^{-4}	23
2	Check Valves	1.2×10^{-4}	19
3	Actuation Channels/Trip Modules	1.2×10^{-4}	19
4	Motor Operated Valves	1.0×10^{-4}	16
5	Circuit Breaker/Contactor	1.0×10^{-4}	16
6	Relief Valve	1.5×10^{-5}	2
7	Control Valve (air operated)	1.1×10^{-5}	2
8	Battery	6.7×10^{-6}	1
9	Turbogenerator	3.1×10^{-6}	1

Table 8. Aging sensitivity of component groups at Calvert Cliffs.

Rank	Type	System	Aging Sensitivity (per reactor year)
1	Check Valve	Auxiliary Feedwater	5.5×10^{-3}
2	Circuit Breaker	Reactor Protection	3.1×10^{-3}
3	Trip Relay	Reactor Protection	2.1×10^{-3}
4	Control Valves (air operated)	Auxiliary Feedwater	1.4×10^{-3}
5	Motor Operated Valves	Auxiliary Feedwater	1.4×10^{-3}
6	Pumps	Auxiliary Feedwater	1.4×10^{-3}
7	Motor Operated Valves	High Pressure ECC	4.5×10^{-4}
8	Motor Operated Valves	Service Water	2.9×10^{-4}
9	Diesel Generators	Emergency Power	1.6×10^{-4}
10	Actuators	Safeguard Actuation	1.6×10^{-4}
11	Pumps	Service Water	1.5×10^{-4}
12	Motor Operated Valves	Low Pressure ECC	9.5×10^{-5}
13	Check Valves	High Pressure ECC	9.4×10^{-5}
14	Check Valves	Low Pressure ECC	8.1×10^{-5}
15	Batteries	Emergency Power	6.5×10^{-5}
16	Pumps	High Pressure ECC	4.7×10^{-5}
17	Room Coolers	Service Water	3.3×10^{-5}
18	Check Valves	Service Water	1.3×10^{-5}
19	Pumps	Low Pressure	1.8×10^{-7}

Table 9. Aging sensitivity of component types at Calvert Cliffs.

Rank	Component Type	Aging Sensitivity	% Contribution
1	Check Valve	5.7×10^{-3}	34
2	Circuit Breaker	3.1×10^{-3}	19
3	Relay/Subchannel	2.1×10^{-3}	13
4	Motor Operated Valve	1.9×10^{-3}	11
5	Control Valve (air operated)	1.7×10^{-3}	10
6	Pump/Turbine Pump	1.6×10^{-3}	9
7	Battery	2.6×10^{-4}	2
8	Diesel Generator	1.6×10^{-4}	1
9	Room Cooler	3.3×10^{-5}	1

Table 10. Aging sensitivity of component groups at Grand Gulf.

Rank	Type	System	Aging Sensitivity
1	Motor Operated Valves	Low Pressure ECC	2.3×10^{-4}
2	Motor Operated Valves	Service Water	1.3×10^{-4}
3	Actuators	Safeguards Actuation	9.9×10^{-5}
4	Pump	Service Water	5.9×10^{-5}
5	Check Valves	Service Water	5.9×10^{-5}
6	Motor Operated Valves	High Pressure ECC	5.4×10^{-5}
7	Check Valves	High Pressure ECC	5.4×10^{-5}
8	Check Valves	Low Pressure ECC	2.8×10^{-5}
9	Batteries	Emergency Power	2.4×10^{-5}
10	Pump	Low Pressure ECC	2.4×10^{-5}
11	Pump/Tubine Pump	High Pressure ECC	1.3×10^{-5}
12	Diesel Generator	Emergency Power	9.5×10^{-6}
13	Relief Valves	Reactor Coolant Pressure Control	2.6×10^{-6}

Table 11. Aging sensitivity of component types at Grand Gulf.

Rank	Type	Aging Sensitivity	% Contribution
1	Motor Operated Valves	4.1×10^{-4}	52
2	Check Valves	1.4×10^{-4}	18
3	Actuators	9.9×10^{-5}	13
4	Pump/Turbine Pump	9.6×10^{-5}	12
5	Batteries	2.4×10^{-5}	3
6	Diesel Generators	9.5×10^{-6}	1
7	Relief Valves	2.6×10^{-6}	1

3.3.4 Combined PWR's

allows us to begin making a differentiation between two basic types of reactors.

This section combines the results of the analysis of the two PWR's to determine an overall PWR ranking. The Grand Gulf results are assumed typical of a BWR since information was only available for one plant.

Table 12 presents the aging sensitivity rankings for component groups at PWR's. These results are obtained by adding the results of the component groups at the two PWR's. Check valves of the auxiliary feedwater system and breakers/contactors and trip relays/trip modules of the reactor protection system have the highest potential risk impact as measured by the aging sensitivity measure.

Table 13 presents the combined results for component types of the two plants. Check valves, circuit breakers/contactors, trip modules/ actuation channels, motor operated valves, pumps, and air operated control valves have the highest values of the aging sensitivity measure.

3.4 <u>Additional Components</u>

In this section we estimate the aging sensitivity measure for three additional component types: the reactor vessel, steam generator tubes, and snubbers using existing PRA's and related studies. The calculations in this section are bounding calculations intended to compare the importance of these components to other components at the plant. Table 14 presents the results of these calculations. The following paragraphs discuss the assumptions and implications of the analyses.

3.4.1 Reactor Vessel

The reactor vessel has the highest potential impact on risk of any component in the plant. PRA's generally make the conservative assumption that a failed reactor vessel results in an uncoolable configuration that leads to core meltdown. The aging impact as measured by the aging sensitivity measure is high compared to the other components in the plant.

3.4.2 Steam Generator Tube

A rupture in a steam generator, as an initiating event, results in a small LOCA and consequently loss of heat removal capability of one steam generator. In this situation, core cooling requirements generally are the operation of the auxiliary feedwater system and at least one high pressure injection pump. Table 15 gives an estimate of the tube aging impact based on the cooling requirement for four plants. Consistent with the aging sensitivity measure definition, these estimates are based on simply adding the conditional failure probabilities of the auxiliary feedwater system and the high pressure injection system. The average value from these four plants is included in Table 14. The potential

Table 12: Aging sensitivity of component groups in PWR's.

Rank	Type	System	Aging Sensitivity
1	Check Valves	Auxiliary Feedwater	5.5×10^{-3}
2	Circuit Breaker/Contractor	Reactor Protection	3.2×10^{-3}
3	Trip Relay/Trip Module	Reactor Protection	2.2×10^{-3}
4	Control Valves (air operated)	Auxiliary Feedwater	1.4×10^{-3}
5	Motor Operated Valves	Auxiliary Feedwater	1.4×10^{-3}
6	Pumps	Auxiliary Feedwater	1.4×10^{-3}
7	Motor Operated Valve	High Pressure ECC	4.7×10^{-4}
8	Motor Operated Valve	Service Water	2.9×10^{-4}
9	Pumps	Service Water	2.6×10^{-4}
10	Actuation Channels	Safeguards Actuation	2.1×10^{-4}
11	Check Valve	Low Pressure ECC	1.8×10^{-4}
12	Motor Operated Valve	Low Pressure ECC	1.7×10^{-4}
13	Turbo Generator/Diesel Generator	Emergency Power	1.6×10^{-4}
14	Check Valve	High Pressure ECC	1.0×10^{-4}
15	Batteries	Emergency Power	7.3×10^{-5}
16	Pumps	High Pressure ECC	5.3×10^{-5}
17	Room Coolers	Service Water	3.3×10^{-5}
18	Pumps	Low Pressure ECC	2.0×10^{-5}
19	Relief Valves	Reactor Coolant Pressure Boundary	1.5×10^{-5}
20	Check Valves	Service Water	1.3×10^{-5}

Table 13. Aging sensitivity of component types in PWR's.

Rank	Type	Aging Sensitivity
1	Check Valves	5.8×10^{-3}
2	Circuit Breaker/Contactor	3.2×10^{-3}
3	Trip Module, Relay/Actuation Channel	2.4×10^{-3}
4	Motor Operated Valves	2.3×10^{-3}
5	Pumps	1.7×10^{-3}
6	Control Valves (air operated)	1.4×10^{-3}
7	Turbo Generator/Diesel Generator	1.6×10^{-4}
8	Batteries	7.3×10^{-5}
9	Room Coolers	3.3×10^{-5}
10	Relief Valves	1.5×10^{-5}

Can these two tables be related to anything we have done or can do with our tables even though $\frac{\partial \lambda_i}{\partial t}$ may not be achievable in our assessment because its just not available in literature

I guess ~~a~~ rudimentary attempt we would put aging mech(s) most contributory next to each component with system discrimination, component type and PWR, BWR discrimination as here given

Table 14. Aging sensitivity measures for selected components.

Component	Aging Sensitivity
Reactor Vessel	1
Steam Generator Tube	3×10^{-3}
Snubber	1.8×10^{-5}

Table 15. Aging sensitivity measure calculations for steam generator tubes.

Plant Name	Cooling Requirements	Aging Sensitivity
ANO	1/2 EFWS 1/3 HPIS	6.5×10^{-4} + 4.0×10^{-4} = 1.1×10^{-3}
Oconee	1/2 AFWS 1/3 HPIS	2.4×10^{-4} + 1.4×10^{-3} = 1.6×10^{-3}
Calvert Cliffs	1/2 AFWS 1/3 HPIS	3.0×10^{-3} + 1.7×10^{-3} = 4.7×10^{-3}
Sequoyah	1/3 AFWS 1/3 HPIS	4.3×10^{-5} + 3.5×10^{-3} = 3.5×10^{-3}

risk impact of steam generator tubes as measured by the aging sensitivity measure is higher than that of the standby components analyzed in Section 3.2.

3.4.3 Snubber

In order to determine the aging impact of snubbers we reviewed the results of the Seismic Safety Margins Research Program (7). The case of snubber failure is specific in that it has been done for the Zion plant based on the information given in Reference (7).

The risk associated with snubber failures is characterized by an increased likelihood of a LOCA induced by an earthquake. The earthquake also degrades the safety system that cools the core in the event of a LOCA. In this situation, it is assumed that snubber failure will result in a large or medium LOCA for any earthquake with a magnitude larger than design basis. The dominant core melt sequences for an earthquake induced LOCA contain failure of the Safety Injection System (SIS) to cool the core. A risk impact of the snubber failure is estimated by the following computation:

$$\frac{\partial R}{\partial q} = \sum_{i=1}^{6} a_i \cdot LOCA_i \cdot SIS_i \qquad (17)$$

where

a_i = The earthquake frequency

$LOCA_i$ = The LOCA probability given an earthquake is in the range of a_i

SIS_i = The probability of SIS failure given an earthquake is in the range of a_i.

The summation is over the six accident sequences identified in Reference (7).

Consistent with the definition of risk impact, the snubber is assumed failed. Since the purpose of the snubber is to prevent piping failure, this implies $LOCA_i = 1$ in Equation (17). Now, using the values of a_i and SIS_i given in Reference (7) the risk impact of the snubber failure is calculated from Equation (17).

Earthquake Frequency, a_i		$LOCA_i$		Conditional SIS Failure Probability	
2.52×10^{-4}	x	1	x	4.7×10^{-2}	+
4.55×10^{-5}	x	1	x	1.2×10^{-1}	+
6.57×10^{-7}	x	1	x	2.6×10^{-1}	+
1.61×10^{-7}	x	1	x	5.0×10^{-1}	+
5.31×10^{-8}	x	1	x	7.5×10^{-1}	+
4.10×10^{-8}	x	1	x	9.9×10^{-1}	$= 1.8 \times 10^{-5}$

Hence

$$\frac{\partial R}{\partial q} = 1.8 \times 10^{-5} \quad \text{per reactor year}$$

If snubbers are tested every year as recommended, then

$$\frac{\partial q}{\partial \lambda} = 1 \quad \text{year}$$

The aging sensitivity measure for snubbers as calculated in this manner is moderately high when compared to the other results in Section 3.2. This calculation is an approximation and subject to high uncertainty. Further, the information used is for only one plant that is not located in a high seismic activity zone. The potential risk significance of snubbers will be very site-dependent in general.

3.5 Limitations and Assumptions

but I don't think a much better way exists.

The analysis performed for this report is limited by the available information as well as time and budgeting constraints. Further, the inherent uncertainties in PRA's are limiting factors in identifying the most important components. The results presented in this section are also subject to the uncertainties inherent in PRA's including component failure data uncertainties, modeling uncertainties, and uncertainties in human actions and response. The particular PRA's utilized to determine the component results did not include treatment of all aspects of risk such as seismic analyses, fires, tornados, etc.

The most important limitations of this study are the limited number of plants analyzed and limiting the scope of components studied to those analyzed in the PRA's. The analysis is limited to the effects of complete failure (loss of function); the effects of degradation are not specifically addressed. Also common-cause failures attributed to aging are not specifically addressed.

This report considers only some of the components that are potentially important to risk. We did not consider components whose primary purpose is to mitigate the consequences of severe accidents such as containment spray nozzles, piping and pumps. The importance to risk of components that mitigate accident consequences is not easy to determine in light of the large uncertainties associated with the phenomenology and fission product behavior of severe accidents. We did not consider structural components such as the containment and containment lining. Piping and wiring are not explicitly considered in these analyses and components such as the reactor vessel, steam generator tubes and snubbers are treated only superficially for example purposes.

4. CONCLUSIONS AND RECOMMENDATIONS

In this section we draw conclusions from the results of the aging sensitivity calculations and make several recommendations for utilization of the results.

4.1 Conclusions

has this been pursued more recently?

The information presented in a standard PRA does not include time dependent effects. In determining the risk level at a plant, PRA's generally use a time averaged unavailability. Aging issues deal with the time dependent nature of risk. This limits the nature of the information that can be extracted from a PRA without extensively modifying the PRA. This report suggests a method for determining the potential risk significance of aging effects that is based on determining the sensitivity of risk to increases in failure rate. This adaptation of PRA results enables us to identify the components that have the most significant impact on risk if their failure rates increase due to aging or service wear effects without describing the time-dependent behavior of the failure rate. The information extracted from PRA's in this manner can be quite useful in guiding research efforts if used in context.

The results of the analysis indicate the most risk significant components at a plant depend on a number of factors including plant system design, testing, and maintenance intervals and operating procedures. The key components with regard to risk can be different at each plant owing to differences in system design or testing, maintenance and operating practices.

Based on the component results in Section 3 many of the potentially most risk significant components are in the auxiliary feedwater system, the reactor protection system and the service water systems. Pumps, check valves, motor operated valves, circuit breakers, and actuating circuits are the component types that have the most potential risk impact based on the aging sensitivity measure. These results must be coupled with time-dependent failure rate characteristics to complete the risk impact due to component aging.

Components not analyzed in PRA's or components assumed to have negligible failure rates can be important to risk if their failure rates increase substantially. Research programs are already in place for some of these components such as the reactor vessel, reactor coolant piping, and steam generator tubes.

4.2 Recommendations

4.2.1 Use of Results

The risk aging sensitivity defined in this report is a measure of the sensitivity of risk to changes in component failure rates. Those components with the highest aging sensitivity cause the greatest impacts on risk if their failure rates increase substantially.

These results are intended to provide guidance to the selection of components for further study and as a guide toward prioritizing resources. Three levels of results are provided. We recommend using the results of the third level (component type rankings) as a ranking of the most important component types. To focus research further we recommend concentrating efforts on a particular component type (such as motor operated valves) or the type of operating environment typical of the systems that have the highest potential impact for that component type (the auxiliary feedwater system for example).

what can we say about this?

These results make no assumptions about which components are most susceptible to aging processes. The significance of a aging mechanism can be obtained by combining the risk aging sensitivity as presented here with estimates of the increase in the time-dependent failure rate. Estimates of time dependent failure rates can be obtained from experimental programs, analytical models or operating history. Ideally, if an equation for time dependent failure rate were obtainable (from an analytical model or a data correlation) the time dependent risk associated with a component can be approximated by:

$$R_i(t) = G_i \cdot \lambda_i(t) \tag{18}$$

where

$R_i(t)$ = The time dependent risk and

$\lambda_i(t)$ = the time dependent failure rate.

The risk increase associated with the aging process could be quantified by integrating Equation (18) over the time period of interest. In practice a good estimate of time dependent failure rate will be difficult to obtain. For prioritization with respect to aging it is sufficient to focus resources on those components that have potentially high impact on risk (as measured by the aging sensitivity measure) and also have failure rates that are most affected by aging and service wear effects (as determined by data, analytical or experimental studies).

Is this what GAL is starting to do?

We recommend limited data or analytical studies for each class of component to determine if any aging or service wear effects are evident from the available data bases. A more extensive analysis can evaluate those components that have a relatively high potential risk significance and exhibit some evidence of age related degradation.

4.2.2 Interfaces

The aging program in general and the risk significance task in particular can benefit from the products of other NRC and industry programs including the Accident Sequence Evaluation Program (ASEP) and the data gathering programs (LER's, NPRDs, and others). The ASEP program is designed to provide analysis of the dominant accident sequences for most LWR's in the United States. As a part of this program the cutsets for the dominant sequences will be identified and risk importance measures will be calculated for a large number of components. When the results are made available it will be possible to apply the methods outlined in this report to a broad range of plants. This will provide a good basis for assigning priorities to component classes based on the risk estimates at a large number of plants rather than the three analyzed here. The approaches used in ASEP will allow identification of the most risk significant components and systems based on plant design and other operating characteristics. This information will assist in making specific recommendations as to what type of inspection and preventive maintenance programs will be most effective in controlling risk at different plants based on plant design.

Has this been done?

4.3 Suggestions for Future Work

The risk aging sensitivity measure identifies the potential risk impact of components in nuclear power plant PRAs. This provides direction for evaluating aging effects; however, there are other important issues that must be addressed to fully understand aging phenomena.

A necessary complement to the risk aging sensitivity measure is a description of the time-dependent effects of aging on component failure rates. Initial estimates of these effects could possibly be estimated from older plant operating history and component failure data. A complete description will include:

(1) Identification of component types that are susceptible to aging

(2) The environmental conditions and system applications that influence component aging

(3) Time-dependent functions defining component failure rates.

This study recommends these factors be investigated first for the components that have high potential risk impact as determined by the risk aging sensitivity measure. Sensitivity calculations employing Weibull type aging functions[8] based on current knowledge of relative material aging rates could further focus this research effort.

Investigation of components that do not appear in PRA dominant cutsets is also necessary. The basic effect of aging phenomena is changes in component failure characteristics. Components now believed non-dominant in PRAs can become major contributors to risk when they are susceptible to significant aging. Identification of sensitive component types and important environmental conditions will provide direction for identifying these components.

Other areas where aging effects can influence risk include:

(1) Common cause failures among components that have similar aging susceptibility

(2) Ability of component testing to detect aging effects

(3) Ability of repair efforts to compensate for age-related deterioration

(4) Aging effects and external events such as earthquakes and floods.

A well-defined effort to investigate these concerns will provide a better understanding of the effects of aging phenomena.

5.0 REFERENCES

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APPENDIX A

AGING SENSITIVITY OF OCONEE COMPONENTS GROUPED BY TYPE AND SYSTEM

Table A-1. Aging sensitivity of Oconee components grouped by type and system.

Component Type	System	Component Designator	Aging Sensitivity (Per Reactor Year)
Pump	LPSW	LPSW-P3B	2.3×10^{-5}
		VP1	2.3×10^{-5}
		LPSW-P3A	3.3×10^{-5}
		VP2	3.3×10^{-5}
	LPIS & ECCR	LP-P1A	1.0×10^{-5}
		LP-P1B	1.0×10^{-5}
	HPIS	HP-1AB	3.2×10^{-7}
		HP-1C	3.9×10^{-6}
	AFWS	EFP-A	3.0×10^{-6}
		EFP-B	3.0×10^{-6}
		EFP-TD	8.7×10^{-8}
Valve			
Motor Operated	LPIS & ECCR	LP-17	1.0×10^{-5}
		LP-18	1.0×10^{-5}
		LP-5	1.0×10^{-5}
		LP-8	1.0×10^{-5}
		LP-22	1.0×10^{-5}
		LP-21	9.0×10^{-6}
	ECCR	LP-19	6.0×10^{-6}
		LP-20	6.0×10^{-6}
	HPIS	HP-24	6.0×10^{-6}
		HP-26	6.0×10^{-6}
		HP-27	4.0×10^{-6}
		HP-25	4.0×10^{-6}
	AFWS	FDW-372	3.0×10^{-6}
		FDS-382	3.0×10^{-6}
		C-156	8.6×10^{-8}
	LPSW & AFWS	LPSW-137	8.6×10^{-8}

Table A-1. Continued

Component Type	System	Component Designator	Aging Sensitivity (Per Reactor Year)
Manual	LPIS & HPIS	LP-28	0
	LPIS & ECCR	LP-11	0
		LP-15	0
		LP-13	0
		LP-16	0
		Test A	0
		Test B	0
	HPIS	HP-101	0
		HP-118	0
		HP-148	0
		HP-114	0
		HP-111	0
	AFWS	C-575	0
		C-576	0
		MS-90	0
		MS-91	0
		FDW-88	0
		C-157	0
Check	LPIS & ECCR	CF-12	1.0×10^{-5}
		CF-14	1.0×10^{-5}
		LP-31	1.0×10^{-5}
		LP-12	1.0×10^{-5}
		LP-48	1.0×10^{-5}
		LP-33	1.0×10^{-5}
		LP-14	1.0×10^{-5}
		LP-47	1.0×10^{-5}
		LP-30	9.0×10^{-6}
		LP-29	9.0×10^{-6}
	AFWS	FDW-232	6.0×10^{-6}
		FDW-317	6.0×10^{-6}
		FDW-233	6.0×10^{-6}

Table A-1. Continued

Component Type	System	Component Designator	Aging Sensitivity (Per Reactor Year)
Check (Continued)	AFWS (Continued)	FDW-319	6.0×10^{-6}
		FDW-373	3.0×10^{-6}
		FDW-370	3.0×10^{-6}
		FDW-383	3.0×10^{-6}
		FDS-380	3.0×10^{-6}
	HPIS	HP-113	4.0×10^{-6}
		HP-102	4.0×10^{-6}
Air Operated	AFWS	FDW-315	6.0×10^{-6}
		FDW-316	6.0×10^{-6}
		MS-93	8.6×10^{-8}
		MS-87	8.6×10^{-8}
		MS-94	8.6×10^{-8}
		MS-95	8.6×10^{-8}
Relief	SRS	Q	1.5×10^{-5}
Contactor	RPS	RPS E	1.3×10^{-5}
		RPS F	1.3×10^{-5}
Circuit Breaker	RPS	CB A	2.6×10^{-5}
		CB B	2.6×10^{-5}
		CB C	1.3×10^{-5}
		CB D	1.3×10^{-5}
Remote Trip Module	RPS	RTM 1	1.3×10^{-5}
		RTM 2	1.3×10^{-5}
		RTM 3	1.3×10^{-5}
		RTM 4	1.3×10^{-5}
Actuation	ESFAS	CH 4	4.3×10^{-5}
		CH 3	1.0×10^{-5}
		CH 1	6.0×10^{-6}
		CH 2	4.0×10^{-6}

Table A-1. Continued

Component Type	System	Component Designator	Aging Sensitivity (Per Reactor Year)
Battery	EPS DC	BAT A	4.0×10^{-6}
		BAT B	4.0×10^{-6}
Turbogenerator	EPS AC	TG 1	2.0×10^{-6}
		TG 2	2.0×10^{-6}

Table A-2. Aging sensitivity of Calvert Cliffs components grouped by type and system.

Component Type	System	Component Designator	Aging Sensitivity (per reactor year)
Pump	AFWS	TP21	6.88×10^{-4}
		TP22	6.88×10^{-4}
	SWS	S22	6.5×10^{-5}
		SW22	6.5×10^{-5}
		CC21	9.4×10^{-7}
		CC22	1.3×10^{-5}
		S21	8.0×10^{-6}
		SW21	3.0×10^{-6}
	HPIS & ECCR	HP21	2.8×10^{-5}
		HP23	1.9×10^{-5}
	LPIS & ECCR	LP22	9.0×10^{-8}
		LP21	9.0×10^{-8}
Valve Motor Operated	AFWS	MOV-4071	6.88×10^{-4}
		MOV-4070	6.88×10^{-4}
	HPIS	MOV-659	1.9×10^{-4}
		MOV-660	1.9×10^{-4}
		MOV-656	1.16×10^{-5}
		MOV-654	9.03×10^{-6}
	SWS	CV-5152	6.45×10^{-5}
		CV-5153	6.45×10^{-5}
		CV-5212	6.45×10^{-5}
		CV-5162	2.84×10^{-5}
		CV-5208	2.84×10^{-5}
		CV-5160	1.1×10^{-5}
		CV-5206	1.1×10^{-5}
		CV-3824	1.1×10^{-5}
		CV-5210	3.0×10^{-6}
		CV-5150	3.0×10^{-6}
	HPIS & LPIS & ECCR	MOV-4143	2.8×10^{-5}
		MOV-4142	2.02×10^{-5}

Table A-2. Continued

Component Type	System	Component Designator	Aging Sensitivity (per reactor year)
Valve Motor Operated (Continued)	ECCR	MOV-4144	1.9×10^{-5}
		MOV-4145	1.4×10^{-5}
	LPIS	CV-657	0
		MOV-658	0
		CV-306	0
Manual	AFWS	C3	0
		C4	0
		P1	0
		P4	0
		S6	0
		P2	0
		P6	0
		S8	0
		H1	0
		H2	0
	SWS	M111	0
		M105	0
		M106	0
		M107	0
		M108	0
		M110	0
		M113	0
		M114	0
		M116	0
		M9A	0
		M28A	0
	HIPS & ECCR	M30	0
		M47	0
		M32	0
		M51	0
	LPIS & ECCR	M34	0
		M54	0
		M55	0
		M28	0

Table A-2. Continued

Component Type	System	Component Designator	Aging Sensitivity (per reactor-year)
Manual (Continued)	LPIS & ECCR (Continued)	M42	0
		M43	0
Air Operated	AFWS	CV-4511	6.88×10^{-4}
		CV-4512	6.88×10^{-4}
Check	AFWS	P3	6.88×10^{-4}
		S5	6.88×10^{-4}
		P5	6.88×10^{-4}
		S7	6.88×10^{-4}
		H5	6.88×10^{-4}
		H6	6.88×10^{-4}
		S3	6.88×10^{-4}
		S4	6.88×10^{-4}
	HPIS & LPIS & ECCR	C65	2.8×10^{-5}
		C66	2.0×10^{-5}
	HPIS & ECCR	C37	2.8×10^{-5}
		C64	2.8×10^{-5}
		C39	1.9×10^{-5}
		C61	1.9×10^{-5}
	ECCR	C21	1.9×10^{-5}
		C20	1.4×10^{-5}
	SWS	C115	1.3×10^{-5}
	LPIS & ECCR	C41	9.0×10^{-8}
		C63	9.0×10^{-8}
		C35	9.0×10^{-8}
		C56	9.0×10^{-8}
Trip Relay	RPS	K1	5.2×10^{-4}
		K2	5.2×10^{-4}
		K3	5.2×10^{-4}
		K4	5.2×10^{-4}

Table A-2. Continued

Component Type	System	Component Designator	Aging Sensitivity (per reactor year)
Circuit Breaker	RPS	1A	3.9×10^{-4}
		2A	3.9×10^{-4}
		3A	3.9×10^{-4}
		4A	3.9×10^{-4}
		1B	3.9×10^{-4}
		2B	3.9×10^{-4}
		3B	3.9×10^{-4}
		4B	3.9×10^{-4}
Actuators	ESFAS (for SWS)	SIB7	8.2×10^{-5}
	(for HPIS)	SIA2	2.8×10^{-5}
	(for HPIS)	SIB2	2.0×10^{-5}
	(for HPIS)	SIA1	1.2×10^{-5}
	(for HPIS)	SIB1	9.0×10^{-6}
	(for ECCR)	RASA1	6.5×10^{-6}
	(for ECCR)	RASB1	4.3×10^{-6}
	(for LPIS)	SIA3	1.0×10^{-7}
	(for LPIS)	SIB3	1.0×10^{-7}
Battery	EPS DC	BAT21	6.5×10^{-5}
		BAT12	9.0×10^{-8}
		BAT22	9.0×10^{-8}
Diesel	EPS AC	D12ST	9.0×10^{-5}
		D21ST	6.5×10^{-5}
Room Cooler	SWS	R21	1.9×10^{-5}
		R22	1.4×10^{-5}

Table A-3. Aging sensitivity of Grand Gulf components grouped by type and system.

Component Type	System	Component Designator	Aging Sensitivity (per reactor year)
Pump	SSWS	C001A-A	2.9×10^{-5}
		C001B-B	2.9×10^{-5}
		C002-C	1.2×10^{-6}
	RHR & LPCIS	C002B-B	1.2×10^{-5}
		C002A-A	1.1×10^{-5}
	RCICS	C001	4.3×10^{-6}
	HPCS	C001-C	2.8×10^{-6}
	LPCIS	C002C-B	6.9×10^{-7}
	LPCS	C001-A	6.0×10^{-7}
Valves			
Motor Operated	SSWS	F001A-A	3.1×10^{-5}
		F001B-B	3.1×10^{-5}
		F005A-A	2.9×10^{-5}
		F005B-B	2.9×10^{-5}
		F018A-A	4.3×10^{-6}
		F018B-B	4.0×10^{-6}
		F011-C	1.2×10^{-6}
	RHR	F014A-A	2.5×10^{-5}
		F068A-A	2.5×10^{-5}
		F014B-B	2.5×10^{-5}
		F068B-B	2.5×10^{-5}
		F003A-A	1.2×10^{-5}
		F047A-A	1.2×10^{-5}
		F003B-B	1.2×10^{-5}
		F047B-B	1.2×10^{-5}
		F024A-A	1.1×10^{-5}
		F024B-B	1.1×10^{-5}
		F048A-A	1.1×10^{-6}
		F048B-B	1.1×10^{-6}
		F087A-A	8.2×10^{-7}
		F052A-A	8.2×10^{-7}
		F026A-A	8.2×10^{-7}
		F087B-B	8.2×10^{-7}

Table A-3. Continued

Component Type	System	Component Designator	Aging Sensitivity (per reactor year)
Valves Motor Operated (Continued)	RHR (Continued)	F052B-B	8.2×10^{-7}
		F026B-B	8.2×10^{-7}
	RHR & LPCIS	F004B-B	1.2×10^{-5}
		F004A-A	1.2×10^{-5}
	RCICS	F013-A	6.0×10^{-6}
		F045-A	6.0×10^{-6}
		F068-A	6.0×10^{-6}
		F010-A	6.0×10^{-6}
		F064-A	6.0×10^{-6}
		F063-B	6.0×10^{-6}
		TTV	6.0×10^{-6}
		TGV	6.0×10^{-6}
	HPCS	F004-C	2.8×10^{-6}
		F001-C	2.8×10^{-6}
	SPMS	F002A-A	1.9×10^{-6}
		F002B-B	1.9×10^{-6}
	LPCIS	F242-B	6.9×10^{-7}
		F004C-B	6.9×10^{-7}
		F042B-B	6.9×10^{-7}
		F027B-B	6.9×10^{-7}
		F042A-A	6.0×10^{-8}
		F027A-A	6.0×10^{-8}
	LPCS	F001-A	6.0×10^{-7}
		F005-A	6.0×10^{-7}
Manual	SSWS	F199A	0
		F199B	0
		F023A	0
		F023B	0
		F185A	0
		F185B	0

Table A-3. Continued

Component Type	System	Component Designator	Aging Sensitivity (per reactor year)
Manual (Continued)	SSWS (Continued)	F186A	0
		F186B	0
		F013	0
	RHR	F130A	0
		F120A	0
		F130B	0
		F120B	0
		F102A	0
		F103A	0
		F102B	0
		F103B	0
		F210A	0
		F210B	0
		F083A	0
		F083B	0
	RHR & LPCIS	F029B	0
		F029A	0
	RCICS	F200	0
		F016	0
	HPC	F205	0
	LPCIS	F239	0
		F029C	0
		F039B	0
		F039A	0
	LPCS	F007	0
Check	SSWS	F008A	2.9×10^{-5}
		F008B	2.9×10^{-5}
		F012	1.2×10^{-6}
	RHR & LPCIS	F031B	1.2×10^{-5}
		F031A	1.1×10^{-5}

Table A-3. Continued

Component Type	System	Component Designation	Aging Sensitivity (per reactor year)
Check (Continued)	RCICS	F040	6.0×10^{-6}
		F066	6.0×10^{-6}
		F065	6.0×10^{-6}
		F204	6.0×10^{-6}
		F011	6.0×10^{-6}
	HPCS	F005	2.8×10^{-6}
		F024	2.8×10^{-6}
		F002	2.8×10^{-6}
	RHR	F054A	8.2×10^{-7}
		F054B	8.2×10^{-7}
	LPCIS	F241	6.9×10^{-7}
		F031C	6.9×10^{-7}
		F041B	6.9×10^{-7}
		F041A	6.0×10^{-8}
	LPCS	F003	6.0×10^{-7}
		F006	6.0×10^{-7}
Relief	SRS	P	2.6×10^{-6}
Turbine	RCICS	C002	6.0×10^{-6}
Actuators	ESFAS (for SSWS)	SAC	2.9×10^{-5}
		SBC	2.9×10^{-5}
		SCC	1.2×10^{-6}
	(for RHR & LPCS & LPCIS)	LRACT	1.4×10^{-5}
	(for RHR & LPCIS)	BCACT	1.3×10^{-5}
	(for RCICS)	RACT	6.0×10^{-6}
	(for HPCS)	HACT	2.8×10^{-6}
	(for SPMS)	SAACC	1.9×10^{-6}
		SBACC	1.9×10^{-6}

Table A-3. Continued

Component Type	System	Component Designator	Aging Sensitivity (per reactor year)
Battery	EPS DC	BATA	1.9×10^{-5}
		BATB	4.0×10^{-6}
		BATC	1.2×10^{-6}
Diesel	EPS AC	DIESEL1	4.3×10^{-6}
		DIESEL2	4.0×10^{-6}
		DIESEL3	1.2×10^{-6}

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13 ABSTRACT (200 words or less)

This report presents a method for focusing additional research on aging phenomena that affects nuclear power plant components. Specifically, the method ranks components using a risk aging sensitivity measure that describes the change in risk due to changes in component failure rate. Describing the aging phenomena and the resulting time-dependent component failure rate changes is beyond the scope of this study.

The applications use average component unavailability equations currently employed in PRAs to calculate the risk aging sensitivity. A more exact calculation is possible by using unavailability equations that include the time-dependent characteristics of component failure rates; however, these time-dependent characteristics are not well-known. The risk aging sensitivity measure presented here is, therefore, segregated from these time-dependent effects and addresses only the time-independent portion of aging phenomena. The results identify the component types that show the most potential for risk change due to aging phenomena. Future research on the time-dependent portion of aging phenomena for these component types is needed to completely describe the risk impact due to component aging.

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