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Prioritization of TIRGALEX - Recommended Components for Further Aging Research

Prepared by I.S. Levy, J. Wreathall, G. DeMoss, A. Wolford,
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ABSTRACT

The "Plan for Integration of Aging and Life Extension," developed by Technical Integration Review Group for Aging and Life Extension (TIRGALEX) in May 1987, identified the safety-related nuclear power plant structures and components (S/C) that should be prioritized for further evaluation by the NRC's Nuclear Plant Aging Research Program (NPAR).

This report documents the results of an expert panel workshop established to perform the S/C prioritization activity. Prioritization was primarily based upon criteria derived from a specially-developed risk-based methodology. This methodology incorporates the effect upon plant risk of both component aging and the effectiveness of current industry aging management practices in mitigating that aging.

An additional set of criteria used to categorize the S/C is the importance of aging research on S/Cs to the resolution of generic safety issues (GSI) and/or to identified NRC/NRR user needs. The resultant S/C categorization was to provide additional information to decision makers, but was not used to calculate final S/C ranks.

ACKNOWLEDGMENTS

This report documents aging-related research activities sponsored by the Office of Nuclear Regulatory Research, Division of Engineering, of the U.S. Nuclear Regulatory Commission. It was funded through the Nuclear Plant Aging Research (NPAR) Program, J. P. Vora, Project Manager and Program Coordinator. These activities could not have been successful without support from a variety of additional sources.

EG&G-Idaho, Inc. [Idaho Nuclear Engineering Laboratory (INEL)], through its NPAR project (managed by P. MacDonald and B. Cook) supported Science Applications International Corporation (SAIC) in the development of the basic Risk Significance of Component Aging (RSCA) model. It also developed the comprehensive component failure data base from which the precalculated aging failure data used in this workshop were derived. David Satterwhite, a co-author of the data base report, graciously provided updated information shortly before the workshop.

Pacific Northwest Laboratory (PNL), through its NPAR project (managed by A. B. Johnson, Jr.) supported the expedited development and application of an enhanced model, the Risk Significance of Component Aging and Aging Management Practices (RSCAAMP) model, for this workshop. SAIC was the major subcontractor to PNL on this project and was responsible for applying the RSCAAMP methodology in the workshop environment. SAIC provided subcontract management and staff support from its Columbus, Ohio, office as well as staff support from its McLean, Virginia, and New York, New York, offices.

Several PNL staff members, through participation in two review activities (a peer review of the RSCAAMP model and a test of the workshop process), provided valuable insights on areas of improvement that contributed to the success of this workshop.

Kathy Wilson, Conference Coordinator at the Battelle Seattle Research Center, and her support staff ensured that the workshop participants had the requisite conference facilities, equipment, and support staff to carry out their work efficiently.

EXECUTIVE SUMMARY

BACKGROUND AND PURPOSE

At the direction of the Executive Director for Operations (EDO), in April 1986 the NRC established the Technical Integration Review Group for Aging and Life Extension (TIRGALEX) to develop a plan to integrate the NRC's aging and life extension activities. In May 1987, TIRGALEX finalized its plan (TIRGALEX 1987); it was approved by the Office Directors of RES, NRR, NMSS, AEOD, and IE and was reviewed by the representative from OGC. The TIRGALEX plan identified the safety-related structures and components (S/Cs) that should be prioritized for subsequent evaluation in the NRC Nuclear Plant Aging Research (NPAR) Program.

The safety-related S/Cs that the TIRGALEX plan identified for evaluation are listed in Table S.1. During the development of the technical information for this workshop, ac/dc buses were found to have sufficient risk importance and were added to the TIRGALEX list. In its deliberations while developing the plan, TIRGALEX specifically chose not to identify systems for potential study, only the components of which the systems are comprised.

Pacific Northwest Laboratory was assigned the responsibility for prioritizing the TIRGALEX components for subsequent evaluation by the NPAR Program.

TABLE S.1. TIRGALEX List of Components[a]

1. Reactor pressure vessel
2. Containment (metal and concrete)
3. Other Category I concrete structures
4. Reactor coolant piping and safe ends
5. Other safety-related piping
6. Steam generator (PWR)[b]
7. Reactor coolant pump casing
8. Pressurizer (PWR)
9. Control rod drive mechanism (CRDM)
10. Cables, connectors, and penetrations
11. Emergency diesel generator
12. Reactor internals
13. RPV[c] support (sliding foot) (PWR)
14. Recirculation piping safe ends (BWR)[d]
15. Snubbers
16. Instruments and controls
17. Switchgear and relays
18. Valves
19. Pumps
20. Motors
21. Turbines
22. Heat exchangers
23. Compressors
24. Fans/chillers
25. Batteries
26. Battery chargers/inverters
27. Transformers
28. Fuel storage racks
29. Accumulators/tanks
30. AC/DC buses[e]

(a) This list was not prioritized by TIRGALEX nor intended to be inclusive. It was anticipated that components would be added if warranted by the prioritization studies.
(b) Pressurized-Water Reactor.
(c) Reactor Pressure Vessel.
(d) Boiling-Water Reactor.
(e) Added since TIRGALEX report as result of analyses prior to the expert panel workshop.

SCOPE

PNL conducted an expert panel workshop to prioritize the TIRGALEX-recommended set of nuclear power plant S/Cs.

Several considerations of the NRC and PNL provided the ground rules within which the workshop was conducted. These were:

- TIRGALEX list of components, with additions as the pre-workshop studies or expert panel analyses may warrant
- aging of current plants (e.g., during their original license period)
- incorporate an understanding of aging and its effects (e.g., define the contribution of S/C aging to plant risk)
- assess the adequacy of current industry practices for managing component aging within acceptable levels of risk
- importance of S/C aging of individual components/component groups on plant risk
- application of the "Risk Significance of Component Aging" (RSCA) methodology (being developed by W. E. Vesely of SAIC under the NPAR Program) to S/C prioritization
- use of operational failure data
- use of expert judgement through an interdisciplinary panel
- importance of aging research on S/Cs to resolution of generic safety issues (GSIs) and identified Office of Nuclear Reactor Regulation (NRR) user needs[a] to aid NRC decision-makers, but not to formally prioritize the components.

The expert panel consisted of the following membership: R. J. Budnitz, Panel Chairman; P. J. Amico; P. L. Appignani; S. H. Bush; L. J. Chockie; S. Kasturi; T. M. Laronge; and D. A. Wesely.

The panel membership represented expertise in a full spectrum of relevant technical areas: PRAs, structures, electrical and mechanical components, component reliability, materials behavior and failure analyses, in-service inspection, operations and maintenance, as well as safety, regulatory, and aging and life extension issues.

(a) "User Need Letter--Nuclear Plant Aging Program" written by H. R. Denton to E. S. Beckjord, dated April 9, 1987.

The expert panel was supplied with the TIRGALEX list of components, prioritization criteria, prioritization methodologies, and technical support material prior to the workshop. The panel used judgement to score the S/Cs for each criterion and to rank the S/Cs relative to one another.

Two sets of prioritization criteria were provided, "risk-based" criteria and "other technical" criteria.

The risk-based criteria were used to assess:

- the potential increase in plant risk from component aging; and
- the adequacy of current aging management practices for maintaining risk at acceptable levels.

The other technical criteria were used to identify:

- Generic Safety Issues (GSIs) that could directly benefit from aging research on an S/C; and
- Identified NRR user needs that could directly benefit from research in the NPAR program on an S/C.

This categorization of S/Cs against these other technical criteria provides additional information to NRC decision-makers, but were not used to rank the S/Cs. An overview of the workshop methodology is presented in Figure S.1.

The outputs from the expert panel workshop were the following:

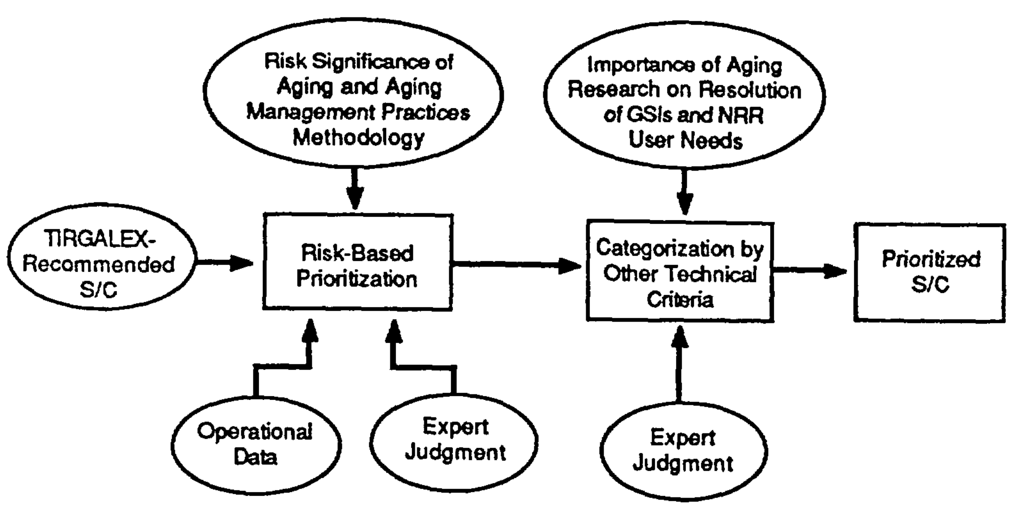

FIGURE S.1. Overview of Workshop Process structure and component prioritization for the risk-based criteria (includes scores for each of the five criteria for each S/C and an integrated ranking of all S/Cs)

structure and component categorization for the other technical criteria (identification of those S/Cs for which aging research would be expected to benefit the resolution of a GSI and/or an identified NRC user need)

The remainder of this executive summary presents highlights in each of these two areas.

PRIORITIZATION OF STRUCTURES AND COMPONENTS USING RISK-BASED CRITERIA

The risk-based criteria were established through the development and application of a state-of-the-art risk-based methodology, the Risk Significance of Component Aging and Aging Management Practices (RSCAAMP) model.

The RSCAAMP model allows the assessment of both the risk significance of component aging and the effectiveness of current industry management practices for maintaining an acceptable plant risk level in the presence of component aging. The RSCAAMP model was developed by enhancing the Risk Significance of Component Aging (RSCA) methodology, which was developed to evaluate a component's contribution to plant risk due to aging (Vesely 1987b, 1987c). In the basic RSCA model, the change in a component's contribution to risk due to aging is a function of the component's importance to risk (N), the rate at which the component's failure rate is increasing due to aging (A), and the interval during which the component is aging (L). The basic methodology has been expanded from a treatment of individual, plant-specific components to a treatment of component groups[a] for generic applications.

Equation (S.1) represents the basic model.

$$\Delta R = N \times A \times \frac{L^2}{2}, \qquad \text{(S.1)}$$

where ΔR = The change in plant risk due to the aging of a component. The risk measure adopted for the S/C prioritization activity is core damage frequency (CD/year).

N = The normal risk importance associated with the component. It is the difference between the core damage frequency calculated in the PRA when the component is always available and the core

(a) In the remainder of this report, we will use "component" to represent any level of component aggregation, whether singular, as appropriate for RPV, or multiple, as in motor operated valves.

damage frequency if the component is defined to be unavailable [e.g. unavailability = 1.0; (ANSI/IEEE 1987)]. This risk importance measure is expressed as the derivative of risk (ΔR) with respect to component unavailability. Hence, the units are the same as those used for the risk measure (e.g., CD/yr).

A = The increase in failure rate (from the rate used in the PRA) due to aging of the component. A is expressed as failures per unit time squared [also termed failure acceleration (Vesely 1987c)]. We use the units $hr^{-1}yr^{-1}$, or the annual increase in hourly failure rate.

L = The interval during which the component is aging (e.g. between overhauls); its units are months.

The product of the normal risk importance (N) and aging failure rate (A) is termed the Core Damage Frequency Acceleration (CDFA):

$$\text{CDFA} = N \times A \ . \qquad \text{(S.2)}$$

CDFA is also described as the "risk significance of aging" (Vesely 1987c).

The basic RSCA methodology applies directly only to PRA-based components. It uses risk importance as data calculated from four PRAs from NRC-funded studies and uses aging failure rates calculated from information in the INEL data base, generated for the NPAR Program.

Precalculated values of N, A, and CDFA were constructed from Equations (S.2) for the PRA-based components. The panel reviewed the PRA S/C list and the risk importances (N) and aging failure rates (A) and made some changes by disaggregating components, applying experience with newer PRAs and judgements on relative failure rates of components.

Structures and components not appearing in PRAs fall into two categories: 1) those primarily intended to prevent core damage accidents and 2) those designed to mitigate and control post-accident releases. Several components, such as the RPV and reactor coolant piping, fall into each category. The panel assigned the structures and components having a dual role to core damage prevention. Solely based on judgment, the panel assigned the N and A values to the non-PRA-based accident prevention components and incorporated them into the existing lists for the PRA-based components. For S/Cs with the major role to mitigate accident consequences, it was necessary to adopt an equivalent risk importance value that reflected the panel's opinion of the overall significance of accident mitigation to safety. The panel judged it appropriate to consider the accident-mitigation S/C (containment), after a core damage accident has occurred, to have a risk importance equal to the highest risk importance for any of the accident-prevention S/Cs. This recognized that the containment provides a barrier to the release of radioactive materials to the public that is of equal importance to the RPV. Based on this assessment, the panel incorporated the N and A values for the accident-mitigation S/Cs into the previously integrated lists for the accident-prevention S/Cs. For this final integrated list of all S/Cs, the risk significance of aging (CDFA) was then calculated.

Enhancements were made to the basic RSCA model so that the effect of current industry aging management practices on plant risk could be evaluated in an expert panel framework. The enhanced model does this by 1) defining an acceptable or control value of the component's aging risk contribution and identifying those components for which risk contribution exceeds this value; 2) defining current industry aging management practices and evaluating their adequacy for maintaining the risk contribution of these aged components within the control value; and 3) calculating the relative contributions to plant risk of aged components given this defined adequacy of industry practices.

An acceptable or control value of ΔR was defined as being ΔR_c. ΔR_c is the limit placed upon the additional contribution to plant risk due to the aging of a component. The control value is established by considering the NRC's safety goal core damage frequency as the measure of total plant risk. Then, a portion of this goal value is allocated to each component such that the sum of the individual component risk contributions will not exceed the goal value. The allocated value for each component is the limit placed upon ΔR_c. The panel chose 1E-7 CD/yr as the working value for ΔR_c.

This control value of component risk contribution was then used to screen from further consideration those components whose risk significance of aging (CDFA) is so low that there is no need to assess the effectiveness of current industry aging management practices upon them. The basis for this screening is the increase in risk that would result if aging of the component were allowed to continue, without overhaul or replacement, for the full 40-year design life of the plant. This screening criterion was applied to the integrated list of PRA-based, non-PRA-based accident prevention, and accident mitigation components, for which the panel had previously developed N and A values (and for which CDFA values were then calculated). A few components were eliminated. The remaining components are with risk significance of aging sufficiently high that adequate aging management practices are required to bring component risk contributions down within ΔR_c.

To assess the adequacy of the current industry aging management practices in controlling the changes in risk due to aging, an acceptable or control overhaul interval, L_c, was defined:

$$L_c = \frac{2 \times \Delta R_c}{CDFA} \tag{S.3}$$

For the panel to evaluate the adequacy of current aging management practices, a value for L that is representative of current industry practice is required. First, L_{act} was defined as the actual (effective) interval between component overhauls that is representative of current industry practice for that component. Next, the ratio of L_{act} to L_c (the control overhaul interval) was defined to represent the adequacy of current practices in controlling risk:

$$\left(\frac{L_{act}}{L_c}\right)^2 = \text{adequacy of current industry aging management practices in controlling risk} \qquad (S.4)$$

The effect of industry aging management practices on the risk contribution of an aged component is represented by:

$$\Delta R = \Delta R_c \times \left(\frac{L_{act}}{L_c}\right)^2 \qquad (S.5)$$

If industry practices are adequate,

$$\left(\frac{L_{act}}{L_c}\right)^2 \leqq 1, \text{ and } \Delta R \leqq \Delta R_c \qquad (S.6)$$

That is, the risk increase due to aging, when aging is adequately managed, remains within an acceptable value.

If industry practices are not adequate,

$$\left(\frac{L_{act}}{L_c}\right)^2 > 1, \text{ and } \Delta R > \Delta R_c \qquad (S.7)$$

That is, the risk increase due to aging, when aging is inadequately managed, exceeds the acceptable value.

Then, L_{act} was defined in terms of parameters that allow the elicitation of quantitative input relative to the adequacy of current aging management practices, and to do so in terms of both the adequacy of aging detection practices and the adequacy of aging mitigation practices.

$$L_{act} = \frac{L_{ind}}{P_D \times P_{R/D}} \qquad (S.8)$$

where L_{act} = The actual (effective) interval, representative of current industry practices, that the component ages without mitigation.

L_{ind} = The surveillance/test interval representative of current industry practices. The panel limited L_{ind} to the surveillance/test interval for the risk significant failure mode (e.g., that used to generate N).

P_D = The probability of successfully detecting aging degradation in the component within the surveillance/test interval. The panel limited this aging degradation to that causing the increased rate of risk significant failures (A), these being for the failure mode used to define N.

$P_{R/D}$ = The probability of successfully mitigating this aging degradation process, given its successful detection in the component (e.g., the aging clock returned to zero or "good as new").

The resultant, fully developed equation for the RSCAAMP model becomes:

$$\Delta R = N \times A \times \frac{1}{2}\left(\frac{L_{ind}}{P_D \times P_{R/D}}\right)^2 \qquad (S.9)$$

The factors N, A, L_{ind}, P_D, and $P_{R/D}$ are used as the S/C prioritization criteria. Each factor is normalized into a scoring scheme, with a score of 5 representing the highest risk effect from that factor and 1 representing the lowest risk effect on a logarithmic scale (see method described in Appendix B). The scores for each S/C are accumulated (addition of logarithms) to obtain a figure of merit (FOM), which represents the ΔR value for that S/C. Final ranking (prioritization) of the S/C is based upon these calculated ΔR values. This ranking incorporates both the risk significance of aging and the effectiveness of industry practices in maintaining aging within an acceptable risk level.

The process used by the expert panel for the risk-based prioritization of the TIRGALEX S/Cs is summarized in Figure S.2. In applying the RSCAAMP model to prioritization of the TIRGALEX components, the expert judgement was used to:

- disaggregate the TIRGALEX S/Cs
- evaluate the pre-calculated values for normal risk importance (N) and failure rate increase due to aging (A) for the PRA-based components
- generate estimates of N and A for the non-PRA-based S/Cs
- generate estimates of the effectiveness of current industry aging management practices (L_{ind}, P_R, and $P_{R/D}$) for all S/Cs
- provide an integrated ranking of the PRA- and non-PRA-based components.

Table S.2 presents the components identified by TIRGALEX, the 4-plant PRAs, and the final list evaluated by the expert panel.

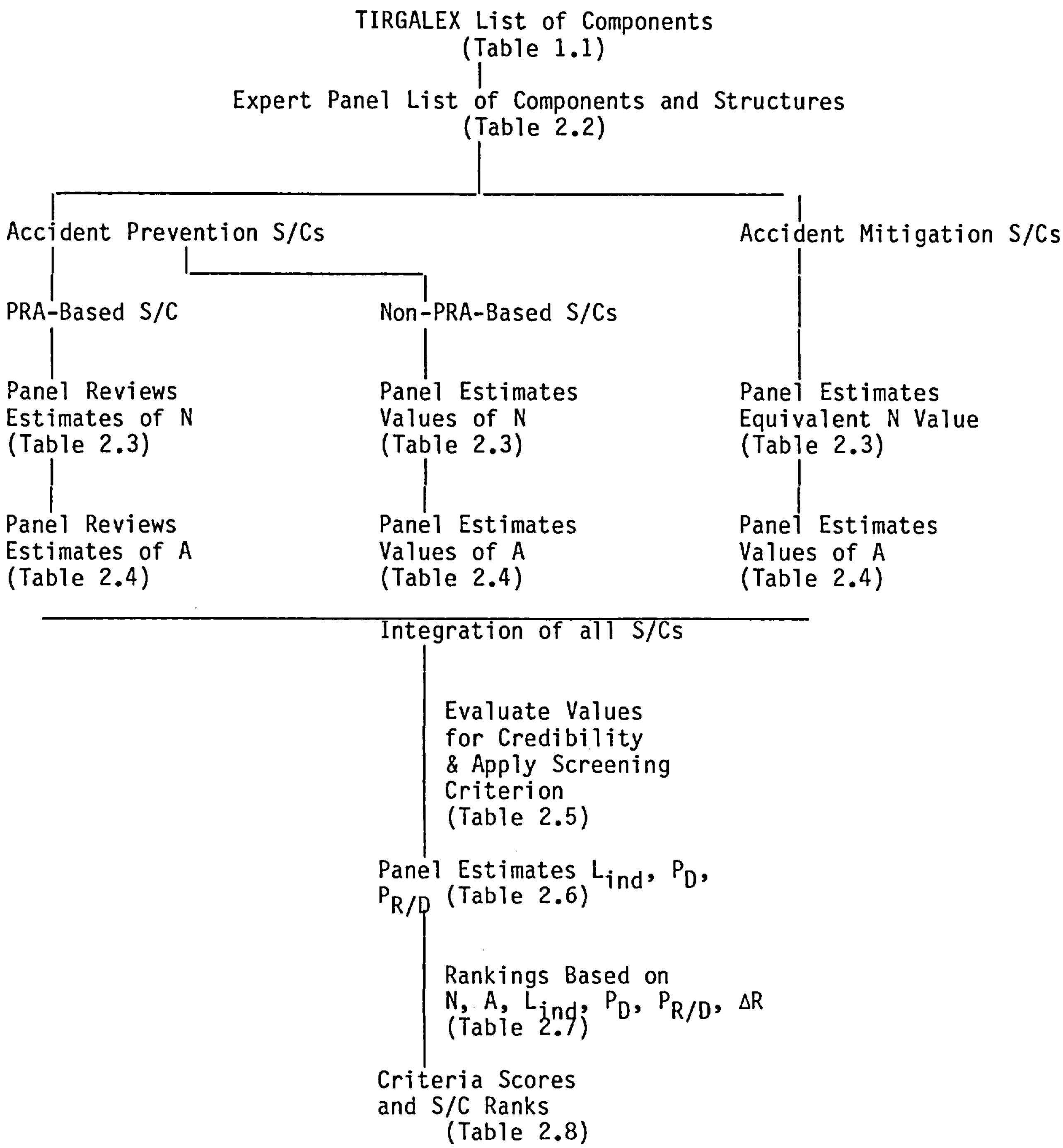

FIGURE S.2. Expert Panel Process for Risk-Based Prioritization

TABLE S.2. Structures/Components Evaluated by Expert Panel

TIRGALEX Groups	4-Plant PRA Components	Components Chosen by the Expert Panel
1. Reactor pressure vessel (RPV)		1. RPV
2. Containment		2. Containment a. BWR b. Other
3. Other Category I structures		3. Other Category I structures
4. Reactor coolant piping (RCP) and safe ends (SE)		4. RCP & SE a. Large LOCA b. Small LOCA, PWR c. Small LOCA, BWR
5. Other safety-related piping (SRP)		5. Other SRP a. Large (10-24 in.) pipe[a] b. Small (6-10.) pipe[b]
6. Steam generator (S/G)		6. Steam generator a. S/G tube b. S/G shell
7. Reactor coolant pump (RCP) casing		7. RCP casing
8. Pressurizer		8. Pressurizer
9. Control rod drive mechanism (CRDM)		9. CRDM a. BWR b. PWR
10. Cables, connectors, and penetrations		10. Cables, connectors, a. Cables b. Connectors (* Penetrations as part of containment)
11. Diesel generator	11. Diesel generator	11. Diesel generator
12. Reactor internals		12. Reactor internals
13. RPV support (PWR)		13. RPV support (PWR)
14. Recirculation piping safe ends	14. Small LOCA, BWR	
15. Snubbers		15. Snubbers
16. Instruments and Controls (I&C)	16. Instruments and Controls (I&C) a. Thermostat	16. Instruments and Controls (I&C) a. Thermostat b. Transfer switch[c] c. Bistable trip[c]
17. Switchgear/relays	17. Switchgear/relays a. Relay (load) b. Circuit breaker c. Transfer switch d. Bistable trip unit	17. Switchgear Relays a. Relay b. Circuit breaker

TABLE S.2. (contd)

TIRGALEX Groups	4-Plant PRA Components	Components Chosen by the Expert Panel
18. Valves	18. Valves a. Air-operated valve (AOV) b. Check valve c. Hydraulic valve d. Manual valve e. Motor-operated valve (MOV) f. Safety/relief valve (S/RV)	18. Valves a. Air-operated valve b. Check valve c. Hydraulic valve d. Manual valve e. Motor operated valve f. Safety/relief valve
19. Pumps	19. Pumps a Motor driven pump b. Turbine driven pump	19. Pumps a. Motor driven pump b. Turbine driven pump
20. Motors		20. Motors (included in valves, pumps, etc.)
21. Turbines	21. Turbines	21. Turbines
22. Heat exchangers	22. Heat exchangers a. Heat exchangers b. Air conditioners	22. Heat exchangers
23. Compressor		23. Compressor(d)
24. Fans/chillers	24. Fan	24. a. Chiller b. Fan
25. Batteries	25. Batteries	25. Batteries
26. Battery chargers/inverters	26. Battery chargers/inverters a. Battery chargers b. Inverters	26. Battery chargers/inverters a. Battery chargers b. Inverters c. Rectifier(e)
27. Transformers	27. Transformers	27. Transformers
28. Fuel storage racks		28. Fuel storage racks
29. Accumulator/tanks	29. Tank	29. Tanks a. Medium pressure tank b. Atmospheric pressure tank c. High pressure tank
30. AC/DC bus	30. AC/DC buses a. AC bus b. DC bus	30. AC/DC buses a. AC bus b. DC bus
		31. Bolts

(a) Represented by service water system piping for which HPI could not keep up if break occurred.
(b) Represented by piping in letdown and reactor water cleanup (RWCU) systems, for which HPI could just keep up if small break occurred.
(c) Moved by panel from S/C-17, 4-Plant PRA.
(d) Compressors in Instrument Air System.
(e) Moved by panel from non-assigned S/C (see Table 2.1).

Table S.3 presents the scores for the prioritization criteria and the final integrated ranking of the S/Cs. The S/Cs are arranged in decreasing order of their contribution to plant risk increase. Those components whose $\Delta R(40)$ values were below 1E-7 CD/yr and were not evaluated for adequacy of aging management practice are included in this table; their rankings are placed at the nominal value of one (1). To stay within the guidelines of considering aging only in the context of the current 40-year license period, the calculations of L_{act} (Equation S.8) were truncated to a value of 480 months.

The impact on component risk contribution of aging and aging management practices, as well as the component's normal risk importance, were evaluated. For the top-ranked components (ranks 5 and 4), all of these factors were significant; no one factor was dominant.

The importance of including component aging and its management in assessing plant risk was dramatized by some unexpected findings. Some components not considered in PRAs were found in the top rank (rank of 5) for risk contribution (cables, connectors, small safety-related piping); conversely, concrete structures and diesels that are considered in PRAs to be significant contributors to plant risk were found in the lower ranks (ranks of 2 and 4, respectively); and, perhaps as significant, is the number of "abundant small components" (e.g., small other safety-related piping, cables, connectors, and S/G tubes) found in the top ranking (four of the five groups ranked 5) when one might intuitively have thought of the major non-redundant, defense-in-depth structures (RPV and containment) to be top-ranked [RPV was ranked 3, containment (other) ranked 2, with only the special case of the BWR-Mk-1 containment (due to aging) ranked 5].

If L_{act} had not been truncated, several components would have L_{act} values well above 480 months and, as a result, would have higher final risk increase values. This impacts the consideration of plant relicensing. If a nuclear plant were allowed to operate beyond 40 years, the ineffectiveness of aging management practices represented by the components' untruncated values would result in continuing and increasing risk. (Note: this assumes no change in those practices.) Of particular concern would be S/C-1: RPV, S/C-2b: containment-other, and S/C-3: other concrete structures since these are permanent structures (though the possibility of RPV replacement has been considered).

Table S.4 presents the current status of component research in the NRC Plant Aging Research Program for the ranked components (from Table S.3).

PRIORITIZATION OF STRUCTURES AND COMPONENTS USING "OTHER TECHNICAL" CRITERIA

TIRGALEX S/Cs were categorized using the other technical criteria, e.g., the importance of aging research on S/Cs to the resolution of GSIs and/or to an identified NRC/NRR user need.

TABLE S.3. Structures/Components Prioritization Criteria Scores and Final Rankings (Truncated to 40 years)

Component	N Score	A Score	L_{ind} Score	P_D Score	$P_{R/D}$ Score	Risk Increase Rank
5.b. Small other safety pipe[a]	2	4	4	5	1	5[b]
10.a. Cables	4	2	4	5	1	5[b]
2.a. Containment (BWR)	5	3	3	1	2	5
10.b. Connectors	3	3	4	4	1	5
6.a. S/G tube	1	5	4	3	4	5
19.b. Turbine pump	3	5	2	3	1	4
17.a. Relay	4	4	1	4	1	4
11. Diesel	3	5	1	3	1	4
12. RX internals	4	2	3	5	1	4
17.b. Breaker	4	2	3	3	1	3
18.e. Motor operated valve	3	5	1	2	1	3
4.c. BWR pipe (small LOCA)	2	3	4	4	1	3
19.a. Motor pump	3	4	2	3	1	3
5.a. Large other safety pipe[c]	3	2	3	5	1	3
16.a. Thermostat	3	3	3	2	1	3
24.a. Chillers	2	4	3	2	1	3
1. RPV	5	1	5	1	5	3[b]
25. Battery	3	4	1	1	1	3
23. Compressor (instr. air)	2	4	1	4	1	3
18.a. Air operated valve	2	4	3	3	1	2
30.b. DC bus	4	1	3	3	1	2
9.a. CRDM (BWR)	4	2	3	1	1	2
18.b. Check valve	2	2	3	5	1	2
24.b. Fan	2	4	3	2	1	2
22. Heat exchanger	3	2	1	5	1	2
31. Bolts	1	4	3	3	1	2
30.a. AC bus	4	1	3	3	1	2
18.f. Safety/relief valve	1	4	3	1	1	2
2.b. Containment (other)	5	1	4	5	4	2[b]
3. Other concrete structures	5	1	4	5	4	2[b]
27. Transformer	3	2	3	2	1	1
26.b. Inverter	1	5	2	3	1	1
16.b. Transfer switch	1	4	3	4	1	1
15. Snubbers	1	5	3	1	1	1
18.c. Hydraulic valve	1	3	3	3	1	1
21. Turbine	1	3	4	1	4	1
16.c. Bistable	1	3	3	2	1	1
18.d. Manual valve	1	2	4	3	3	1
26.a. Battery charger	1	3	2	1	1	1
29.b. Tank (atmos. pres.)	3	1	2	5	2	1
26.c. Rectifier	1	3	2	3	1	1
29.a. Tank (medium pres.)	3	1	2	5	2	1
9.b. CRDM (PWR)	2	1	3	3	1	1
						[d]
8. Pressurizer	1	1	--	--	--	1
6.b. S/G shell	1	1	--	--	--	1
29.c. Tank (high pres.)	1	1	--	--	--	1
4.a. RC P & SE large (LOCA)	1	1	--	--	--	1
7. RC P casing	1	1	--	--	--	1
13. RPV support	1	1	--	--	--	1
28. Fuel rack	1	1	--	--	--	1
4.b. PWR pipe (small LOCA)	2	1	--	--	--	1

(a) 6-10 in. pipe represented by letdown and RWCV systems (see Table S.2).
(b) L_{act} for these components was truncated to a maximum of 40 years (480 months).
(c) 10-24 in. pipe represented by service water system piping (see Table S.2).
(d) Components below this line were eliminated from further consideration by screening criterion.

TABLE S.4. Status of Aging Research on Ranked Components

Component	ΔR Rank	Research on-going	Components of Interest But Not In Scope
5.b. Small other safety pipe[a]	5	x	
10.a. Cables	5	x	
2.a. Containment (BWR)	5	0[b]	
10.b. Connectors	5	x	
6.a. S/G tube	5	--[c]	
19.b. Turbine pump	4	x	
17.a. Relay	4	x	
11. Diesel	4	x	
12. RX internals	4	0	
17.b. Breaker	3	x	
18.e. Motor operated valve	3	x	
4.c. BWR pipe (small LOCA)	3	x	
19.a. Motor pump	3	x	
5.a. Large other safety pipe[d]	3	x	
16.a. Thermostat	3	0	
24.a. Chillers	3		x
1. RPV	3	x	
25. Battery	3	x	
23. Compressor (instr. air)	3	x	
18.a. Air operated valve	2		x
30.b. DC bus	2	x	
9.a. CRDM (BWR)	2	x	
18.b. Check valve	2	x	
24.b. Fan	2		x
22. Heat exchanger	2	x	
31. Bolts	2	0	
30.a. AC bus	2	x	
18.f. Safety/relief valve	2		x
2.b. Containment (other)	2	0	
3. Other concrete structures	2		x
27. Transformer	1	x	
26.b. Inverter	1	x	
16.b. Transfer switch	1	0	
15. Snubbers	1	x	
18.c. Hydraulic valve	1	0	
21. Turbine	1	0	
16.c. Bistable	1	x	
18.d. Manual valve	1	0	
26.a. Battery charger	1	x	
29.b. Tank (atmos. pres.)	1	0	
26.c. Rectifier	1	x	
29.a. Tank (medium pres.)	1	0	
9.b. CRDM (PWR)	1	x	
			[e]
8. Pressurizer	1	0	
6.b. S/G shell	1	0	
29.c. Tank (high pres.)	1		x
4.a. RCP & SE large (LOCA)	1	x	
7. RCP casing	1	0	
13. RPV support	1	x	
28. Fuel rack	1	0	
4.b. PWR pipe (small LOCA)	1	x	

(a) 6-10 in. pipe represented by letdown and RWCV systems (see Table S.2).
(b) 0: no activity or plan.
(c) Completed FY87.
(d) 10-24 in. pipe represented by service water system piping (see Table S.2).
(e) Components below this line were eliminated from full evaluation by the screening criterion.

The list of potentially important safety issues comes from two sources: 1) a list of GSIs with elements of aging that may benefit from NPAR results; and 2) additional issues discovered as a result of a pre-workshop review of all GSIs.

The TIRGALEX S/Cs were initially screened to identify those S/Cs for which these GSIs could directly benefit from aging research. This prescreening took account of the technical content and available schedules associated with the resolution of the GSIs and with the NRC Plant Aging Research Programs (Vora 1987). Where there was a direct technical connection (i.e., aging was involved in the issue) and the time scales appeared to be compatible, aging research on the S/Cs was identified as potentially having a direct benefit. On the other hand, some S/Cs were clearly not associated with the list of GSIs, aging was not directly related to the issue, or time scales were incompatible with their resolution; aging research on these S/Cs was classified as not beneficial. All GSI evaluations were presented to the panel for discussion and final categorization as whether aging research would benefit or not benefit the resolution of the GSIs. A few GSIs considered relevant, but which were not identified in the current GSI resolution schedule, were also presented to the panel for consideration.

NRR user needs were expressed in the "User Need Letter". In the letter, NRR expressed the need to "... know not only the effects of aging on structures, systems and components, but also the risk significance..." of the process. The "User Need Letter" was reviewed for components deemed to be important by NRR.

In the panel's judgement, many of the generic issues that deal with equipment performance were expected to directly benefit from aging research (schedule permitting), and that further aging research on the components identified from the user needs letter would be of benefit to NRR. Table S.5 shows the components' ranking based on risk, S/Cs for which aging research would benefit the resolution of a GSI and/or a NRC/NRR user need and those S/Cs for which current aging research is not already on-going. Only for Rx internals (meeting GSI and user needs criteria), pressurizer (GSIs), and bolts (GSI) and hydraulic and manual valves (GSI and user needs) is aging research not already on-going.

CONCLUSIONS AND RECOMMENDATIONS

The expert panel workshop was conducted to prioritize the TIRGALEX set of nuclear power plant S/Cs for further evaluation within the NRC Nuclear Plant Aging Research Program. The prioritization was primarily based upon risk-based criteria; other technical criteria were used to categorize those S/Cs for which aging research would benefit the resolution of GSIs and/or identified NRC/NRR user needs but were not used to rank the S/Cs.

From the S/C prioritization using risk-based criteria, the major conclusions from the workshop are the following: 1) the prioritization of S/Cs was accomplished by an expert panel using the multi-factor RSCAAMP methodology. Analysis of the results showed that all the factors of this methodology are

TABLE S.5. Status of Aging Research on S/Cs Ranked by Risk Importance and Other Technical Criteria

	Component	ΔR Rank	Aging Research Important to Resolution of: GSI	Aging Research Important to Resolution of: User Need	Aging Research Not On-Going or of Current Interest
5.b.	Small other safety pipe[a]	5			
10.a.	Cables	5	x,x[b]		
2.a.	Containment (BWR)	5			0
10.b.	Connectors	5	x,x		
6.	S/G tube	5	x		(c)
19.b.	Turbine pump	4	x		
17.a.	Relay	4	x,x,x		
11.	Diesel	4	x	x	
12.	RX internals	4	x		0
17.b.	Breaker	3	x,x		
18.e.	Motor operated valve	3	x,x	x,x,x	
4.c.	BWR pipe (small LOCA)	3			
19.a.	Motor pump	3	x	x	
5.a.	Large other safety pipe[d]	3			
16.a.	Thermostat	3			
24.a.	Chillers	3			
1.	RPV	3	x,x,x	x	
25.	Battery	3		x	
23.	Compressor (instr. air)	3			
18.a.	Air operated valve	2	x	x	
30.b.	DC bus	2	x,x		
9.a.	CRDM (BWR)	2			
18.b.	Check valve	2	x,x	x	
24.b.	Fan	2			
22.	Heat exchanger	2	x		
31.	Bolts	2	x		0
30.a.	AC bus	2	x,x		
18.f.	Safety/relief valve	2	x	x,x	
2.b.	Containment (other)	2			
3.	Other concrete structures	2	x		
27.	Transformer	1			
26.b.	Inverter	1		x	
16.b.	Transfer switch	1			
15.	Snubbers	1	x		
18.c.	Hydraulic valve	1	x	x	0
21.	Turbine	1			0
16.c.	Bistable	1			
18.d.	Manual valve	1	x,x	x	0
26.a.	Battery charger	1		x	
29.b.	Tank (atmos. pres.)	1			0
26.c.	Rectifier	1		x	
29.a.	Tank (medium pres.)	1			0
9.b.	CRDM (PWR)	1			
					(e)
8.	Pressurizer	1	x		0
6.b.	S/G shell	1			0
29.c.	Tank (high pres.)	1			
4.a.	RCP & SE large (LOCA)	1	x,x		
7.	RCP casing	1			0
13.	RPV support	1			
28.	Fuel rack	1			0
4.b.	PWR pipe (small LOCA)	1			

(a) 6-10 in. pipe represented by letdown and RWCV systems (see Table S.2).
(b) More than one "x" indicates multiple GSI and/or user need entries.
(c) Completed FY87.
(d) 10-24 in. pipe represented by service water system piping (see Table S.2).
(e) Components below this line were eliminated from full evaluation by the risk-based screening criterion.

equally important to an assessment of the relative risk importances of aged components. Further, the importance of including component aging and its management in assessing plant risk was dramatized by the unexpected findings in the top-ranked components. 2) Current aging research warrants reevaluation: many low-ranked S/Cs (ranks 1 and 2) are under study and others are considered to be of interest; two of the components in the top two ranks (ranks 5 and 4) are not currently being studied in the Plant Aging Research Program [containment (BWR-Mk-1) and RX internals]. 3) The panel's deliberations and findings highlighted the need for improved PRA and aging-failure data bases, the shortfalls in current industry practices for detecting aging for risk-significant failure modes in components and structures, and the usefulness of the methodology for focusing of research and regulatory actions and for providing to utilities areas to address in order to reduce the risk-contribution of their aged components. 4) This study should be viewed as a starting point for which expert opinion was required; from this study an appropriately-focused research program can generate the data needed for a definitive answer on the relative risk-importance of aged components. 5) The explicit evaluation of the importance of aging-induced common-cause failures would be a useful follow-up exercise to perform; with minor development, the RSCAAMP model would be amenable to such an evaluation.

From the S/C categorization using other technical criteria, the major conclusions from the workshop are as follows: 1) the categorization of S/Cs using the other technical criteria was accomplished satisfactorily. 2) There are many S/Cs for which an aging research program could have direct benefit to the resolution of GSIs. 3) There are fewer, but still a substantial number of S/Cs for which an aging research program could respond directly to identified user needs. 4) Only for a few of these components is aging research not already on-going.

CONTENTS

FIGURES

TABLES

GLOSSARY

AFW - auxiliary feedwater
AOV - air operated valve
ASME - American Society of Mechanical Engineers
ATWS - anticipated transient without scram
BWR - boiling-water reactor
CDFA - core damage frequency acceleration
CRDM - control rod drive mechanism
FOM - figure of merit
GSI - generic safety issues
HPI - high pressure injection
HPCI - high pressure coolant injection
HX - heat exchanger
I&C - instruments and controls
INEL - Idaho Nuclear Engineering Laboratory
IREP - Integrated Reliability Evaluation Program
LOCA - loss-of-coolant accident
LOSP - loss of system pressure
MOV - motor operated valve
NDE - nondestructive examination
NPAR - Nuclear Plant Aging Research
NPP - nuclear power plant
NPRDS - nuclear plant reliability data system
NRC - U.S. Nuclear Regulatory Commission
NRR - Nuclear Reactor Regulation
PNL - Pacific Northwest Laboratory
PORV - power operated relief valve
PRA - probabilistic risk assessment
PWR - pressurized-water reactor
RCIC - reactor core isolation cooling
RCP - reactor coolant pump
RES - Office of Nuclear Regulatory Research
RHR - residual heat removal
RPV - reactor pressure vessel
RSCA - Risk Significance of Component Aging
RSCAAMP - Risk Significance of Component Aging and Aging Management Practices
RSCAAMP factors:

- ΔR - the change in plant risk due to aging of a component
- ΔR_c - the acceptable or control value of ΔR
- N - the normal risk importance associated with a component
- A - the increase in failure rate due to aging of the component
- CDFA - core damage frequency acceleration (NxA)
- L - the interval during which the component is aging
- L_c - the acceptable or control overhaul interval
- L_{act} - the actual (effective) interval between overhauls that is representative of current industry practice for a component

$\left(\frac{L_{act}}{L_c}\right)^2$ - the adequacy of current aging management practices

L_{ind} - the surveillance/test interval representative of current industry practices

$(L_{ind})_{acc}$ - the acceptable industry surveillance or test interval

P_D - the probability of successfully detecting component aging degradation within the surveillance or test interval

$P_{R/D}$ - the probability of successfully mitigating aging degradation, given its successful detection in a component

RSSMAP - Reactor Safety Study Methodology Application Program

RWCU - reactor water cleanup

SAIC - Science Applications International Corporation

S/C - structure/component

SE - safe ends

S/G - steam generator

SRP - safety related piping

S/RV - safety/relief valve

TIRGALEX - Technical Integration Review Group for Aging and Life Extension

1.0 INTRODUCTION

This report documents the scope, methods, results, technical insights, conclusions, and recommendations of an expert panel workshop that was held on November 17-19, 1987. The purpose of the workshop was to prioritize nuclear power plant structures and components (S/Cs) for further evaluation within the Nuclear Plant Aging Research Program being conducted by the Division of Engineering, Office of Nuclear Regulatory Research (RES) of the U.S. Nuclear Regulatory Commission (NRC). The workshop was conducted under the auspices of the NRC's Nuclear Plant Aging Research (NPAR) Program by Pacific Northwest Laboratory (PNL)[a] and Science Applications International Corporation (SAIC), under subcontract to PNL. It was held at the Battelle Seattle Conference Center, a part of the Battelle Seattle Research Center.

For the prioritization process, a state-of-the-art risk-based methodology, the latest compilation of component aging failure data, and expert judgement were utilized to assess the risk impacts of component aging and of industry practices for managing component aging.

1.1 BACKGROUND AND PURPOSE

At the direction of the Executive Director for Operations (EDO), in April 1986 the NRC established the Technical Integration Review Group for Aging and Life Extension (TIRGALEX) to develop a plan to integrate the NRC's aging and life extension activities. In May 1987, TIRGALEX finalized its plan (TIRGALEX 1987); it was approved by the Office Directors of RES, NRR, NMSS, AEOD, and IE and was reviewed by the representative from OGC. The TIRGALEX plan identified the safety-related structures and components that should be prioritized for subsequent evaluation in the NRC Nuclear Plant Aging Research (NPAR) Program.

The safety-related S/Cs that the TIRGALEX plan identified for evaluation are listed in Table 1.1. During the development of the technical information for this workshop, ac/dc buses were found to have sufficient risk importance and were added to the TIRGALEX list. In its deliberations while developing the plan, TIRGALEX specifically chose not to identify systems for potential study, only the components of which the systems are comprised.

Pacific Northwest Laboratory was assigned the responsibility for prioritizing the TIRGALEX components for subsequent evaluation by the NPAR Program. An expert panel workshop approach was used.

(a) Operated for the U.S. Department of Energy (DOE) by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830.

TABLE 1.1. TIRGALEX List of Components[a]

1. Reactor pressure vessel	16. Instruments and controls
2. Containment (metal and concrete)	17. Switchgear and relays
3. Other Category I concrete structures	18. Valves
4. Reactor coolant piping and safe ends	19. Pumps
5. Other safety-related piping	20. Motors
6. Steam generator (PWR)[b]	21. Turbines
7. Reactor coolant pump casing	22. Heat exchangers
8. Pressurizer (PWR)	23. Compressors
9. Control rod drive mechanism (CRDM)	24. Fans/chillers
10. Cables, connectors, and penetrations	25. Batteries
11. Emergency diesel generator	26. Battery chargers/inverters
12. Reactor internals	27. Transformers
13. RPV[c] support (sliding foot) (PWR)	28. Fuel storage racks
14. Recirculation piping safe ends (BWR)[d]	29. Accumulators/tanks
15. Snubbers	30. AC/DC buses[e]

(a) This list was not prioritized by TIRGALEX nor intended to be inclusive. It was anticipated that components would be added if warranted by the prioritization studies.
(b) Pressurized-Water Reactor.
(c) Reactor Pressure Vessel.
(d) Boiling-Water Reactor.
(e) Added since TIRGALEX report as result of analyses prior to the expert panel workshop.

1.2 SCOPE OF THE EXPERT PANEL WORKSHOP

This section describes the ground rules within which the workshop was conducted, the process used by the expert panel, and the expected outputs of the workshop.

1.2.1 Ground Rules

Several considerations of the NRC and PNL provided the ground rules within which the workshop was conducted. These were:

- TIRGALEX list of components, with additions as the pre-workshop studies or expert panel analyses may warrant

- aging[a] of current plants (e.g., during their original license period)
- incorporate an understanding of aging and its effects (e.g., define the contribution of S/C aging to plant risk)
- assess the adequacy of current industry practices for managing component aging within acceptable levels of risk
- importance of S/C aging of individual components/component groups on plant risk
- application of the "Risk Significance of Component Aging" (RSCA) methodology (being developed by W. E. Vesely of SAIC under the NPAR Program) to S/C prioritization
- use of operational failure data
- use of expert judgement through an interdisciplinary panel
- importance of aging research on S/Cs to resolution of generic safety issues (GSIs) and an identified Office of Nuclear Reactor Regulation (NRR) user needs to aid NRC decision-makers, but not to formally prioritize the components.

1.2.2 Process

The expert panel was supplied with the TIRGALEX list of components, prioritization criteria, prioritization methodologies, and technical support material. The panel used their expert judgement to score the S/Cs for each criterion and to rank the S/Cs relative to one another.

Two sets of prioritization criteria were provided, "risk-based" criteria and "other technical" criteria.

1.2.2.1 Risk-based Criteria

The risk-based criteria have two important factors as their bases:

- the potential increase in plant risk from component aging

(a) The TIRGALEX plan defined aging as the cumulative degradation occurring within a component, structure, or system that, if unchecked, may result in loss of function and impairment of safety. Aging may be caused by natural internal chemical or physical processes; external stressors and environment; service wear (cycling, vibration); testing; or improper installation, application, or maintenance.

- the adequacy of current aging management practices for maintaining risk at acceptable levels.

Five risk-based criteria were used to implement these two bases:

N : normal risk importance of the S/C[a]

A : increase in S/C failure rate due to aging

L_{ind} : surveillance/test interval represented by current industry practice

P_D : probability of successfully detecting aging degradation within L_{ind}

$P_{R/D}$: probability of successfully mitigating aging given successful detection.

These risk-based criteria were developed by enhancing the Risk Significance of Component Aging (RSCA) model (Vesely 1987c) for application to this prioritization activity. The enhancement was accomplished through discussions between PNL and SAIC. The RSCA model had already encompassed the risk importance of components (N), the effect on risk of the increase in S/C failure rate due to aging (A), as well as the interval (L) during which aging was continuing unmitigated. The enhancements were to allow for consideration of the effectiveness of current industry practices for managing aging degradation and to do so in a manner that would 1) subdivide those practices into both detection and mitigation practices and 2) easily permit input of expert judgement (L_{ind}, P_D, and $P_{R/D}$). The enhanced model is termed "Risk Significance of Component Aging and Aging Management Practices" (RSCAAMP).

In applying the RSCAAMP model to prioritization of the TIRGALEX components, the expert judgement was used to:

- evaluate the pre-calculated values for normal risk importance (N) and failure rate increase due to aging (A) for the PRA-based components
- generate estimates of N and A for the non-PRA-based S/Cs
- generate estimates of the effectiveness of current industry aging management practices (L_{ind}, P_R, and $P_{R/D}$) for all S/Cs
- provide an integrated ranking of the PRA- and non-PRA-based components.

(a) This is equivalent to the Birnbaum risk importance measure as discussed by W. E. Veseley and T. C. Davis in Evaluation and Utilization of Risk Importance.

1.2.2.2 Other Technical Criteria

In addition to the risk-based criteria, the importance of performing aging research on the S/Cs to the resolution of GSIs and a set of the NRC/NRR user needs identified in "User Need Letter--Nuclear Plant Aging Program." (See Appendices F and G, respectively.)

The S/Cs were screened prior to the workshop to identify those for which GSIs could directly benefit from aging research. These were submitted to the panel for approval.

A similar approach was followed for the relevance of aging research on S/Cs to the identified NRR user needs. The S/Cs were screened for those of direct benefit, and then presented to the panel for approval.

This categorization of S/Cs against these other technical criteria provides additional information to NRC decision-makers, but is not used to rank the S/Cs (per method in Telford et al. 1986). An overview of the workshop methodology is presented in Figure 1.1.

1.2.3 Outputs

The outputs from the expert panel workshop were anticipated to be the following:

- structure and component prioritization for the risk-based criteria (includes scores for each of the four criteria for each S/C and an integrated ranking of all S/Cs)

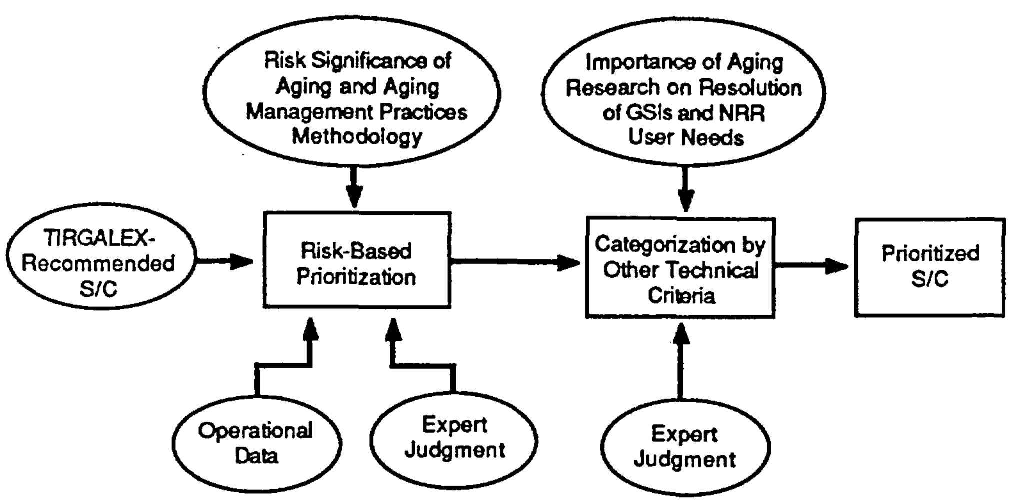

FIGURE 1.1. Overview of Workshop Process

- structure and component categorization for the other technical criteria (identification of those S/Cs for which aging research would be expected to benefit the resolution of a GSI and/or an identified NRC user need)

- technical insights of the panel.

1.3 COMPOSITION OF EXPERT PANEL WORKSHOP

1.3.1 Expert Panel Membership

The expert panel consisted of the following membership: R. J. Budnitz, Panel Chairman; P. J. Amico; P. L. Appignani; S. H. Bush; L. J. Chockie; S. Kasturi; T. M. Laronge; and D. A. Wesely.

The panel membership represented expertise in a full spectrum of relevant technical areas: PRAs, structures, electrical and mechanical components, component reliability, materials behavior and failure analyses, in-service inspection, operations and maintenance, as well as safety, regulatory, and aging and life extension issues.

1.3.2 Workshop Technical Support Staff Membership

Technical participation and workshop facilitation was provided by the workshop technical support staff, which included the following membership:

- co-technical chairman: I. S. Levy - PNL Project Leader, "NRC Plant Aging Research Prioritization Task"

- co-technical chairman: J. Wreathall - SAIC Project Leader, "Applications of Risk Significance of Aging Methodology to NPP Component Prioritization Subtask"

- workshop leader: D. L. Brenchley (PNL)

- methodology support: A. J. Wolford (consultant to SAIC)
G. M. DeMoss (SAIC)

- rapporteurs: E. Collins (SAIC)
D. Jarrell (PNL)

In addition, the workshop had three observers representing the NRC Nuclear Plant Aging Research Program; these were M. Vagins, Branch Chief, Electrical and Mechanical Components Branch, NRC/RES.; A. B. Johnson, Jr., PNL NPAR Project Manager; and B. Cook, EG&G-Idaho (INEL) NPAR Project Manager.

1.4 ORGANIZATION OF THIS REPORT

This report of the expert panel workshop is organized in the following manner:

- Section 2.0 discusses the prioritization of the TIRGALEX components using the "risk-based" criteria. This section discusses the RSCAAMP methodology, the data sources used by the RSCAAMP methodology, the pre-calculated values for the PRA-based components, the application of the RSCAAMP methodology to the non-PRA-based structures and components, and the method for determining the adequacy of current industry aging management practices. The section continues with the findings and their rationale from the prioritization process, the technical insights produced by the panel during its deliberations, and conclusions from the process.

- Section 3.0 discusses the categorization of structures and components using the other technical criteria, e.g., the importance of aging research on S/Cs to resolution of GSI and/or to an identified NRC/NRR user need. Structure/component categorization for each of these two criteria is discussed separately and includes the methodologies used, the results of the processes, comments produced by the panel during its deliberations, and conclusions.

- Section 4.0 presents the overall conclusions and recommendations of the workshop.

2.0 PRIORITIZATION OF STRUCTURES AND COMPONENTS USING RISK-BASED CRITERIA

The purpose of this section is to describe the Risk Significance of Component Aging and Aging Management Practices (RSCAAMP) model and its application by an expert panel tasked with prioritizing the TIRGALEX S/Cs for further evaluation by the NRC Plant Aging Research Program. The data sources used to precalculate values for PRA-based components are identified, the data vulnerabilities are identified, and the data are displayed. The prioritization process described in Section 1.2.2.1 is followed; the results of the interim and final rankings are tabulated, and the rationale for the findings and the major comments and insights of the expert panel are documented.

2.1 THE RISK SIGNIFICANCE OF COMPONENT AGING AND AGING MANAGEMENT PRACTICES METHODOLOGY AND ITS APPLICATION TO S/C PRIORITIZATION

The RSCAAMP model allows the assessment of both the risk significance of component aging and the effectiveness of current industry management practices for maintaining an acceptable plant risk level in the presence of component aging. The model and its application to prioritization of PRA-based S/C and to non-PRA-based S/Cs is described below. The application of the basic model for assessing the risk significance of component aging is first discussed. Then, we discuss the enhancements made to the basic model to allow an assessment of the adequacy of current industry management practices to control aging.

2.1.1 Risk Significance of Component Aging

2.1.1.1 Basic RSCA Model and Definitions

The basic RSCA methodology was developed to evaluate a component's contribution to plant risk due to aging (Vesely 1987b, 1987c). In this model, the change in a component's contribution to risk due to aging is a function of the component's importance to risk (N), the rate at which the component's failure rate is increasing due to aging (A), and the interval during which the component is aging (L). The basic methodology has been expanded from a treatment of individual, plant-specific components to a treatment of component groups[a] for generic applications, as described in Appendix A. This basic model is shown in Figure 2.1.

Equation (2.1), repeated from Figure 2.1, represents the basic model. The derivation for the equation is included in Appendix B:

$$\Delta R = N \times A \times \frac{L^2}{2}, \qquad (2.1)$$

(a) In the remainder of this report, we will use "component" to represent any level of component aggregation, whether singular, as appropriate for RPV, or multiple, as in motor operated valves.

The change in a component's contribution to risk due to aging	=	The component's normal risk importance	X	The increase in the component's failure rate due to aging	X	A factor representing the interval during which the component is aging (e.g. between overhauls)

$$\Delta R = N \times A \times \frac{L^2}{2}$$

FIGURE 2.1. Basic Model for the Risk Significance of Component Aging (RSCA)

where ΔR = The change in plant risk due to the aging of a component. The risk measure adopted for the S/C prioritization activity is core damage frequency (CD/year).

N = The normal risk importance associated with the component. It is the difference between the core damage frequency calculated in the PRA when the component is always available and the core damage frequency if the component is defined to be unavailable [e.g. unavailability = 1.0; (ANSI/IEEE 1987)].[a] This risk importance measure is expressed as the derivative of risk (ΔR) with respect to component unavailability. Hence, the units are the same as those used for the risk measure (e.g., CD/yr).

A = The increase in failure rate (from the rate used in the PRA) due to aging of the component. A is expressed as failures per unit time squared [also termed failure acceleration (Vesely 1987c)]. We use the units $hr^{-1}yr^{-1}$, or the annual increase in hourly failure rate.[b] The panel limited A to the aging mechanism that contributed to the risk significant failure mode, e.g., the one used in the PRAs from which N is generated.

L = The interval during which the component is aging (e.g. between overhauls); its units are months.[b]

Vesely (1987a) has shown that an estimate of the aging rate (A) may be obtained by the method of moments.

The product of the normal risk importance (N) and aging failure rate (A) is termed the Core Damage Frequency Acceleration (CDFA):

(a) Multiple failures of similar or dissimilar components in redundant systems, even in the same cut set, are not explicitly considered in the current prioritization activity.

(b) To convert to the units for ΔR (CD/yr), A is multiplied by 8760 hrs yr^{-1} and L is multiplied by 12 yrs^{-1}.

$$CDFA = N \times A . \tag{2.2}$$

CDFA is also described as the "risk significance of aging" (Vesely 1987c).

Precalculated values of N, A, and CDFA were produced for the PRA-based components and presented to the panel for their review. These data and their sources are presented in the following section.

2.1.1.2 Precalculated Data for the PRA-Based Components

Data Sources. The basic RSCA methodology applies directly only to PRA-based components. It uses risk importance data calculated from four PRAs from NRC-funded studies and uses aging failure rates calculated from information in the INEL data base, generated for the NPAR Program. These data, which support the calculations of N, A, and CDFA, are in Appendix B.

Four PRAs were selected to contribute to the generic risk importances (N) needed for this study. These PRAs were selected because they were under study in the NPAR program's RSCA effort funded through INEL; the plants represent three of the four U.S. nuclear power plant vendors. The four PRAs are for:

- ANO-1 Unit 1: An Integrated Reliability Evaluation Program (IREP) PRA performed on a PWR built by Babcock and Wilcox (B&W). (NUREC/CR-2787)
- Grand Gulf: A Reactor Safety Study Methodology Application Program (RSSMAP) PRA performed on a BWR built by General Electric (GE). (Hatch 1981)
- Calvert Cliffs: A RSSMAP PRA performed on a PWR built by Combustion Engineering. (Kolb 1981b)
- Oconee: A RSSMAP PRA performed on a PWR built by Babcock and Wilcox. (Kolb 1981a)

The components considered in each of the four plant PRAs and their relationship to the TIRGALEX list of S/Cs is shown in Table 2.1. Not all of the TIRGALEX S/Cs are considered by these PRAs. However, the panel was not restricted to the components considered in the PRAs, and, in fact, the panel disaggregated existing component groups and added new components as well. As noted in Section 1.2.2.1, expert judgment is required to develop the N and A values for the non-PRA-based components, and is used to evaluate the pre-calculated values for the PRA-based components.

The number of components contributing to each PRA is shown in Appendix B (by component group in Table B.1 and by system in Table B.2). The component risk importances from these PRAs are shown in Table B.3. These importance values were calculated by SAIC. To provide the generic importance of component groups, medians are taken for the importances for that component group.

TABLE 2.1. Components in the Four-Plant PRA Studies Compared to TIRGALEX List of Components

Component		Calvert Cliffs	Grand Gulf	Oconee	ANO-1
TIRGALEX	4-Plant Study				
1. Reactor pressure vessel					
2. Containment					
3. Concrete structures					
4. RCP and safe ends					
5. Other safety-related piping					
6. Steam generator					
7. RC pump casing					
8. Pressurizer					
9. CRD mechanism					
10. Cables, connectors and penetrations					
11. Emergency diesel gen.	Diesel generators	X	X		X
12. Reactor internals					
13. RPV support					
14. Recirc. piping					
15. Snubbers					
16. Instruments and controls					
17. Switchgear and relays	Circuit breakers	X		X	X
	Contactors			X	
	Relays	X			
	Transfer switch				X
	Bistable				X
18. Valves	Check	X	X	X	X
	Motor operated	X	X	X	X
	Air operated	X		X	
	Manual	X	X	X	X
	Hydraulic				X
	Safety/relief			X	X
	Solenoid				
19. Pumps	Motor driven	X	X	X	X
	Turbine driven	X		X	X
20. Motors					
21. Turbines			X		
22. Heat exchange	Air conditioner	X			X
	Heat exchanger				X
23. Compressors					
24. Fans/chillers	Fans				X
	Chillers				
25. Batteries	Batteries	X	X	X	X
26. Battery chargers/ inverters	Battery charger				X
	Inverter				X
27. Transformers					X
28. Fuel storage racks					
29. Accumulators	Others				
	Bus-ac				X
	Bus-dc				X
	Rectifier				X
	Turbogenerator	X			

While the median values for risk importances were presented to the panel, the individual importances from each of the four PRAs were also made available (as shown in Table B.3). Systems importances and the importances of the components within systems were also calculated and provided to the panel (Table B.4). (Only the importances of components within the systems were used by the panel.) A S/C risk importance value was selected by the panel. In the cases where the panel made changes from the precalculated values, it was usually based on consideration of newer PRAs within their experience.

As shown in Equation (11) of Appendix A, the increase in failure rate due to aging (A) is a function of the aging fraction (f), the mean time-to-aging failure (T_A), and the component failure rates (λ). The aging fractions were taken from Meale and Sutterwhite (1987), the large-scale aging data analysis performed by INEL using the nuclear plant reliability data system (NPRDs) data base. For that study, experts from INEL evaluated the causes of large numbers of component failures and then classified the failures to determine the fraction caused by age-related stressors.

These data were generated from a review of more than 2000 NPRDS records for three vendors [B&W, Westinghouse Electric Corporation (WEC), GE] and fifteen safety related systems. The systems are identified in Table B.5. The supporting data on aging failure rates are presented in Tables B.5 through B.8, including failure rates of components within systems.

The mean time-to-aging failures (T_A) (Table B.5) also comes from Meale and Sutterwhite (1987). The aging fractions (f) are shown in Table B.6; they were calculated for the various safety systems that the data allowed. When no significant differences between systems were noted, the fractions were actually calculated at the aggregate level. These quantities are from straight-forward calculations from the NPRDS data. The failure rates (Table B.7) are best estimate values from various sources. The calculated increase in failure rates due to aging (A) and failure rate doubling times are also presented in this table. The panel relied heavily on the doubling times. For components not represented in the data base, the panel developed their own estimates of aging failure rates. Sometimes these panel estimates were based on comparison with values for other components, or a positioning of the components based on perceived relative aging rates.

Precalculated Values. Precalculated values of N, A, and CDFA were constructed from Equations (2.1), (12) in Appendix A, and (2.2), respectively, for the PRA-based components and from the data sources described above; they are presented in Table C.1. The data source for N is from the generic value in Table B.3; the source for A is Table B.7 which in turn, is, based on data found in Tables B.5 and B.6.

The data for N, A, and CDFA in Table C.1 are arranged by TIRGALEX component number and name. Where the TIRGALEX component represents a group of components for which the PRAs provide subgroupings, these are identified by a, b.

2.1.1.3 Application to Non-PRA-Based Components

Structures and components not appearing in PRAs fall into two categories: 1) those primarily intended to prevent core damage accidents and 2) those designed to mitigate and control post-accident releases. Several components, such as the RPV, fall into each category.

The following TIRGALEX S/Cs (with their respective number from Table 1.1) have an impact on the prevention of core damage but are not included in the PRAs used in this study:

1. Reactor pressure vessel
3. Other Category I concrete structures
4. Reactor coolant piping and safe ends
5. Other safety-related piping
6. Steam generators
7. Reactor coolant pump casing
8. Pressurizer
9. Control rod drive mechanism
10. Cables, connectors, and penetrations
12. Reactor internals
13. RPV support (sliding foot) (PWR)
14. Recirculation piping, safe ends (BWR)
15. Snubbers
16. Instruments and controls

The following TIRGALEX S/Cs have an impact on the mitigation of releases from core damage accidents.

1. Reactor pressure vessel[a]
2. Containment (metal and concrete)
3. Other Category 1 concrete structures[a]
4. Reactor coolant piping and safe ends[a]
5. Other safety-related piping[a]
7. Reactor coolant pump casing[a]
13. RPV support (sliding foot) (PWR)[a]
14. Recirculation piping, safe ends (BWR)[a]
28. Fuel storage racks[b]

For the structures and components having a dual role, the panel assigned them to core damage prevention. This left only the containment to be considered by the panel as a structure used only for accident mitigation.

Solely based on judgment, the panel assigned the N and A values to the non-PRA-based accident prevention components and incorporated them into the

(a) Associated with both core damage prevention and accident mitigation.
(b) Fuel storage racks prevent fuel damage accidents; they were arbitrarily classified with the core damage prevention components.

existing lists for the PRA-based components. Next, the panel performed the equivalent estimations for the structures associated with accident mitigation. The difference between the accident prevention and mitigation S/Cs is in the interpretation of risk importance. As usually evaluated in PRAs, the risk importance parameter indicates the sensitivity of the core damage frequency to increases in the component failure probability. For S/Cs whose major role is the mitigation of accident consequences, it was necessary to adopt an equivalent risk importance value that reflected the panel's opinion of the overall significance of accident mitigation to safety. It was recognized that there is no uniquely correct interpretation of relative importance of accident prevention and accident mitigation, and that this is a matter of continuing NRC policy evolution. At this point, the panel judged it appropriate to consider the accident-mitigation S/C (containment), after a core damage accident has occurred, to have a risk importance equal to the highest risk importance for any of the accident-prevention S/Cs. This recognized that the containment provides a barrier to the release of radioactive materials to the public that is of equal importance to the RPV. Based on this assessment, the panel incorporated the N and A values for the accident-mitigation S/Cs into the previously integrated lists for the accident-prevention S/Cs. For this final integrated list of all S/Cs, the risk significance of aging (CDFA) was then calculated. Next, the task of assessing the effectiveness of current industry aging management practices was undertaken.

2.1.2 Adequacy of Current Aging Management Practices

Enhancements were made to the basic RSCA model so that the effect of current industry aging management practices on plant risk could be evaluated in an expert panel framework.

Aging management practices have the potential for adequate control of aging-enhanced plant risk. The enhancements contained within the RSCAAMP model, described below, model this concept and allow for direct input of quantitative, expert opinion to evaluate the effectiveness of aging management practices. The model does this by 1) defining an acceptable or control value of the component's aging risk contribution and identifying those components whose risk contribution exceeds this value; 2) defining current industry aging management practices and evaluating their adequacy for maintaining the risk contribution of these aged components within the control value; and 3) calculating the relative contributions to plant risk of aged components given this defined adequacy of industry practices.

2.1.2.1 Acceptable (Control) Value Factors for Component Risk Contribution

We define an acceptable or control value of ΔR as being ΔR_c. ΔR_c is the limit we place upon the additional contribution to plant risk due to the aging of a component. The control value is established by considering the NRC's safety goal core damage frequency as the measure of total plant risk. Then, a portion of this goal value is allocated to each component such that the sum of the individual component risk contributions will not exceed the goal value.

Based on this, the panel chose 1E-7 CD/yr as the working value for ΔR_c. The allocated value for each component is the limit placed upon ΔR_c.

Having this control value of component risk contribution, we can screen from further consideration those components with risk significance of aging (CDFA) so low that there is no need to assess the effectiveness of current industry aging management practices. The basis for this screening is the increase in risk that would result if aging of the component were allowed to continue, without overhaul or replacement, for the full 40-year design life of the plant. This would be a conservative estimate for most components; the panel used this basis for screening. From Equations (2.1) and (2.2), the calculated increase in plant risk for this case (e.g., that the screening criterion is $\Delta R(40) < 1E\text{-}7$ CD/yr) is:

$$\Delta R = N \times A \times \frac{L^2}{2} = \text{CDFA} \times \frac{40^2}{2} < 1\ E\text{-}7\ \text{CD/yr} < \Delta R_c \qquad (2.3)$$

Equation (2.3) was applied to the integrated list of PRA-based, non-PRA-based accident prevention, and accident mitigation components, for which the panel had previously developed N and A values (and for which CDFA values were then calculated). A few components were eliminated. The remaining components are those with risk significance of aging sufficiently high that adequate aging management practices are required to bring component risk contributions down within ΔR_c.

To assess the adequacy[a] of the current industry aging management practices in controlling the changes in risk due to aging, we continue with the following steps. With a control value defined for a component's risk contribution, ΔR_c, we now solve Equation (2.3) for an acceptable or control overhaul interval, L_c. It can be shown that:

$$L_c = \sqrt{\frac{2 \times \Delta R_c}{\text{CDFA}}} \qquad (2.4)$$

The control overhaul interval L_c may be viewed as a theoretical value arrived at by applying desirable risk control and is the limit we place upon the component's aging interval with no mitigating action being taken (no refurbishment or overhaul). Equation (2.3) may now be expressed in terms of ΔR_c and L_c:

$$\Delta R_c = \text{CDFA} \times \frac{L_c^2}{2} \qquad (2.5)$$

(a) "Adequacy" refers to the theoretical potential for limiting the changes in risk due to aging through its detection and mitigation.

2.1.2.2 Adequacy of Current Industry Practices Factors

For the panel to evaluate the adequacy of current aging management practices, a value for L that is representative of current industry practice is required. First, we define L_{act} as the actual (effective) interval between component overhauls that is representative of current industry practice for that component. Next, we define the ratio of L_{act} to L_c (the control overhaul interval) to represent the adequacy of current practices in controlling risk:

$$\left(\frac{L_{act}}{L_c}\right)^2 = \text{adequacy of current industry aging management practices in controlling risk} \tag{2.6}$$

The effect of industry aging management practices on the risk contribution of an aged component is represented by:

$$\Delta R = \Delta R_c \times \left(\frac{L_{act}}{L_c}\right)^2 \tag{2.7}$$

From this equation, it may be observed that the adequacy of industry's aging management practices has considerable impact on the risk contribution of aged components since it enters the risk equation raised to the power of two.

If industry practices are adequate,

$$\left(\frac{L_{act}}{L_c}\right)^2 \leqq 1, \text{ and } \Delta R \leqq \Delta R_c \tag{2.8}$$

That is, the risk increase due to aging, when aging is adequately managed, remains within an acceptable value.

If industry practices are not adequate,

$$\left(\frac{L_{act}}{L_c}\right)^2 > 1, \text{ and } \Delta R > \Delta R_c \tag{2.9}$$

That is, the risk increase due to aging, when aging is inadequately managed, exceeds the acceptable value.

We now define L_{act} in terms of parameters that will allow us to elicit quantitative input relative to the adequacy of current aging management practices, and to do so in terms of both the adequacy of aging detection practices and the adequacy of aging mitigation practices.

$$L_{act} = \frac{L_{ind}}{P_D \times P_{R/D}} \tag{2.10}$$

where L_{act} = The actual (effective) interval, representative of current industry practices, that the component ages without mitigation.

L_{ind} = The surveillance/test interval representative of current industry practices. The panel limited L_{ind} to the surveillance/test interval for the risk significant failure mode (e.g., that used to generate N).

P_D = The probability of successfully detecting aging degradation in the component within the surveillance/test interval. The panel limited this aging degradation to that causing the increased rate of risk significant failures (A), these being for the failure mode used to define N.

$P_{R/D}$ = The probability of successfully mitigating this aging degradation process, given its successful detection in the component (e.g., the aging clock returned to zero or "good as new").

Substituting Equation (2.10) into (2.7), the effect of detection and mitigation practices on a component's aging risk contribution is:

$$\Delta R = \Delta R_C \times \left(\frac{1}{L_c} \times \frac{L_{ind}}{P_D \times P_{R/D}} \right)^2 \qquad (2.11)$$

To elicit the panel's expert opinion on the adequacy of industry detection and mitigation practices, we posed the following three questions:

1. What is the representative industry interval for inspection/testing for the risk significant failure mode of the component (L_{ind})?

2. What is the probability that the inspection/test method used will identify the existence of the risk significant aging mechanism (P_D)?

3. Given successful detection, what is the probability of successful mitigation (repair, replacement) of the risk significant aging degradation in the component ($P_{R/D}$)?

L_{act} was calculated from the panel's responses to these questions, which were input to Equation (2.10).

We can now quantify the adequacy of current industry aging management in controlling risk practices by the detection and mitigation of component aging. The adequacy of current industry practices for detecting aging degradation is:

$$A_D = \left(\frac{L_{ind}}{P_D} \times \frac{1}{L_c} \right)^2 \qquad (2.12)$$

The adequacy of current industry practices for mitigating aging, given successful detection, is:

$$A_{R/D} = P_{R/D} \quad , \tag{2.13}$$

where we are stipulating that given successful detection in the interval L_{ind}, the component is immediately shut down (e.g., in accordance with relevant regulations) and repairs are performed such that the aged component no longer contributes to plant risk.

From Equations (2.8), (2.10), (2.12) and (2.13), we can also specify an acceptable industry surveillance/test interval, (L_{ind}) as:

$$(L_{ind})_{acc} \leq (P_D \times P_{R/D}) \times L_c$$

or

$$(L_{ind})_{acc} \leq P_D \times L_c \tag{2.14}$$

given the stipulation with regard to immediate component shutdown upon successful detection of aging noted above.

This relationship shows that, as the probability of detection of aging degradation decreases, the industry surveillance/test (S/T) interval (L_{ind}) must also decrease (e.g., frequency of S/T increase) in order to stay within the control interval (L_c) that defines acceptable risk (ΔR_c).

An illustration of the RSCAAMP model [Equation (2.11)] is presented in Figure 2.2. In this figure, the risk contribution (ΔR) of a component is plotted against the time during which the component is aging. We've plotted time as increments of L_c, the control overhaul interval, and have set the value of L_c at 1 month. Two cases are shown. In the first case, we set $L_{act} = 1$. L_c is not exceeded, that is, industry practices are adequate, and as a result, ΔR_c is not exceeded. In the second case, we set $L_{ind} = 0.25$ month, and both P_D and $P_{R/D} = 0.25$. Thus, $L_{act} = 4$; and L_c is exceeded. That is, industry practices are not adequate; as a result, ΔR_c is exceeded.

2.1.2.3 Relative Contribution to Plant Risk of Aged Components

The fully developed equation for the RSCAAMP model becomes:

$$\Delta R = N \times A \times \frac{1}{2} \left(\frac{L_{ind}}{P_D \times P_{R/D}} \right)^2 \tag{2.15}$$

This model is illustrated in Figure 2.3.

The factors N, A, L_{ind}, P_D, and $P_{R/D}$ are used as the S/C prioritization criteria. Each factor is normalized into a scoring scheme, with a score of 5 representing the highest risk effect from that factor and 1 representing the lowest risk effect on a logarithmic scale (see method described in Appendix B). The scores for each S/C are accumulated (addition of logarithms) to obtain a figure of merit (FOM), which represents the ΔR value for that S/C. Final

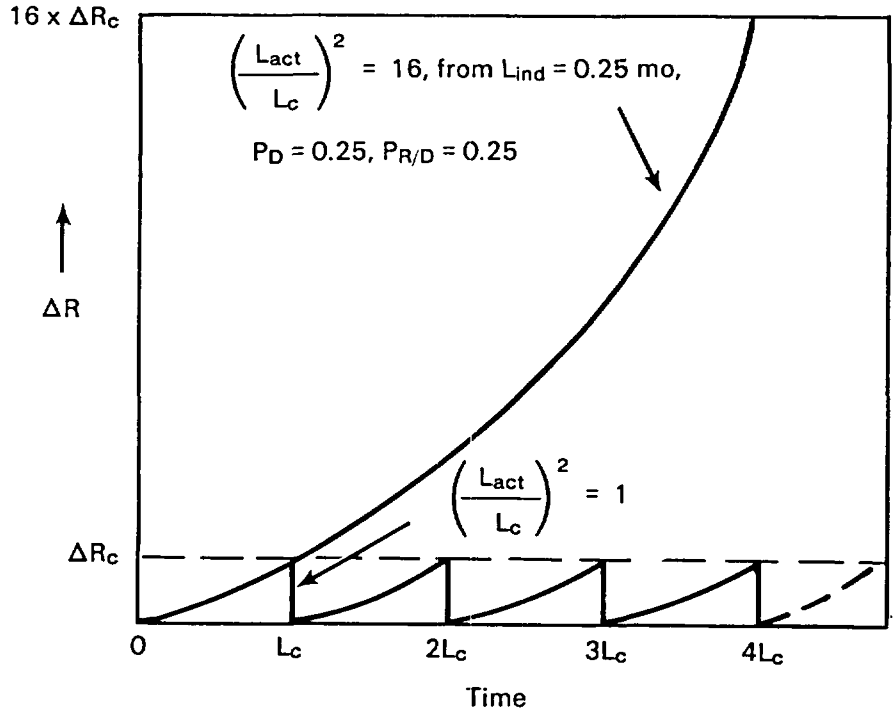

FIGURE 2.2. Effect of Industry Aging Management Practices on Component Aging Risk Contribution [Equation (2.11)]

The change in a component's contribution to risk due to aging	=	The component's normal risk importance	X	The increase in the component's failure rate due to aging	X	A factor representing the effectiveness of industry practices in managing component aging [the actual (effective) interval between overhauls]

$$\Delta R \quad = \quad N \quad X \quad A \quad X \quad \frac{1}{2} \times \left(\frac{L_{ind}}{P_D \times P_{R/D}} \right)^2$$

FIGURE 2.3. The Risk Significance of Component Aging Management Practices Model

ranking (prioritization) of the S/C is based upon these calculated ΔR values. This ranking incorporates both the risk significance of aging and the effectiveness of industry practices in maintaining aging within an acceptable risk level.

The process used by the expert panel for the risk-based prioritization of the TIRGALEX S/Cs is summarized in Figure 2.4. Because many of the specific numerical values generated for the risk based factors are the calculated result, not the cause, of the relative positioning of the components, these numerical values do not have a sufficient basis to justify their use to perform analyses for which the numerical value is a necessary and dominant part of the answer.

2.2 PANEL FINDINGS AND TECHNICAL RATIONALE

The panel's findings together with their rationale, are presented in the following sections. While the broad rationale are discussed with the findings, detailed rationale are provided in Appendix D.

2.2.1 Final Component Selection

The panel reviewed the TIRGALEX component list as it was amplified by the 4-Plant PRA study, which disaggregated some component groups and added S/C-30: ac/dc buses. During deliberations on N and A, the panel developed further disaggregation of the existing groups; added a new group (S/C-31: bolts); and deleted from further consideration the component group, S/C-20: motors. This group was viewed by the panel as redundant since motors are contained within the groups valves, pumps, etc., (as in motor-operated valves), and should not be decoupled from the evaluation of the driven components.

Table 2.2 presents the components identified by TIRGALEX, the 4-plant PRAs, and the final list evaluated by the expert panel. The panel's disaggregation of some groups is evident. The panel's rationale for this consisted of the following:

- S/C-2: containment: to permit consideration of aging of BWR-MK-I containments.

- S/C-4: reactor coolant piping and safe-ends: to permit consideration of variations in risk importance between large- and small-break loss-of-coolant accidents (LOCA) and of aging by stress corrosion cracking that occurs in BWR piping.

- S/C-5: other safety-related piping: to permit consideration of a major failure in large piping (6 in.-24 in.), represented by the important service water system, and a break in smaller piping (6 in.-10 in.) such that high pressure injection (HPI) could just keep up with the coolant loss.

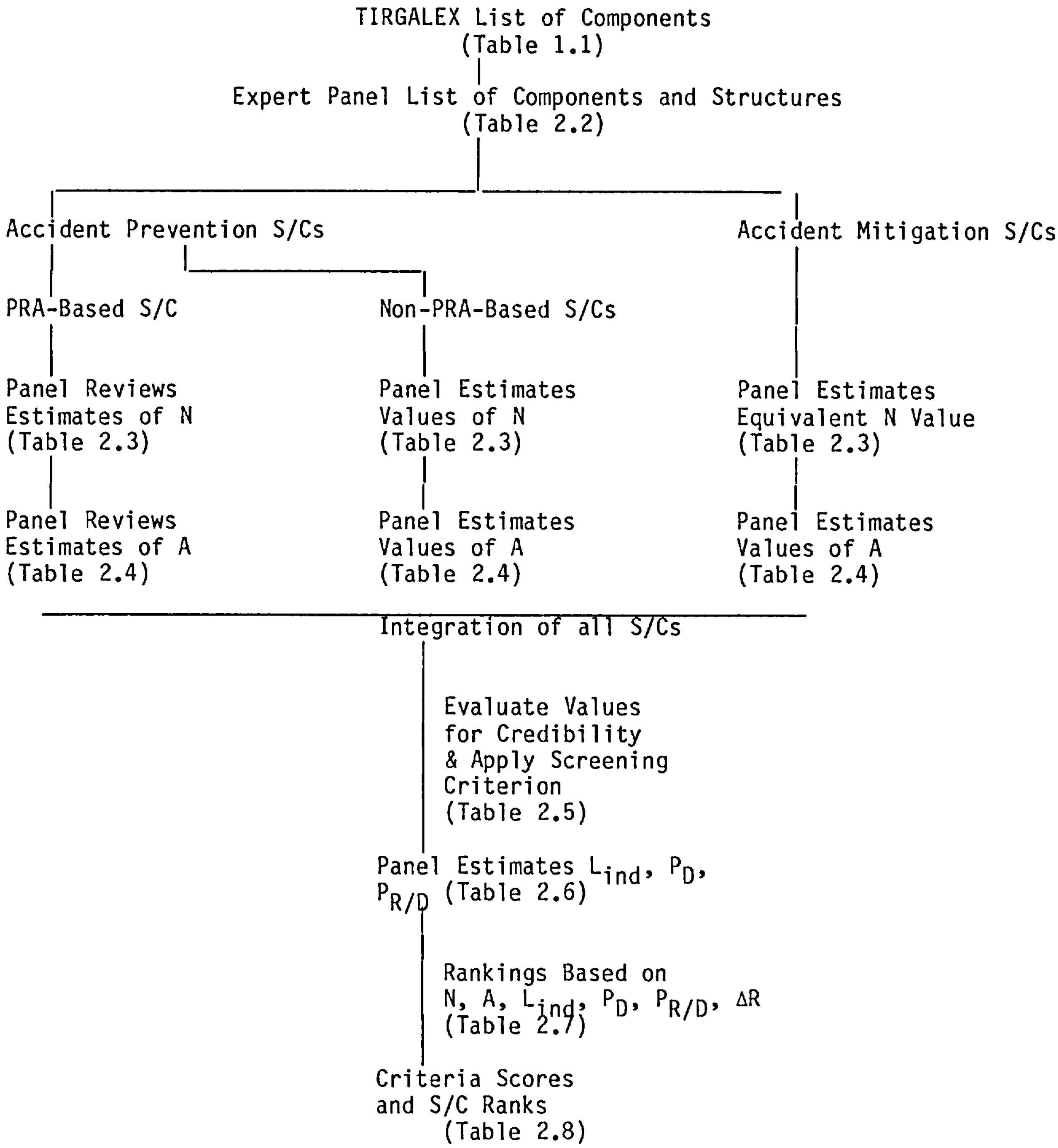

FIGURE 2.4. Expert Panel Process for Risk-Based Prioritization

TABLE 2.2. Structures/Components Evaluated by Expert Panel

TIRGALEX Groups	4-Plant PRA Components	Components Chosen by the Expert Panel
1. Reactor pressure vessel (RPV)		1. RPV
2. Containment		2. Containment a. BWR b. Other
3. Other Category I structures		3. Other Category I structures
4. Reactor coolant piping (RCP) and safe ends (SE)		4. RCP & SE a. Large LOCA b. Small LOCA, PWR c. Small LOCA, BWR
5. Other safety-related piping (SRP)		5. Other SRP a. Large (10-24 in.) pipe[a] b. Small (6-10.) pipe[b]
6. Steam generator (S/G)		6. Steam generator a. S/G tube b. S/G shell
7. Reactor coolant pump (RCP) casing		7. RCP casing
8. Pressurizer		8. Pressurizer
9. Control rod drive mechanism (CRDM)		9. CRDM a. BWR b. PWR
10. Cables, connectors, and penetrations		10. Cables, connectors, a. Cables b. Connectors (* Penetrations as part of containment)
11. Diesel generator	11. Diesel generator	11. Diesel generator
12. Reactor internals		12. Reactor internals
13. RPV support (PWR)		13. RPV support (PWR)
14. Recirculation piping safe ends	14. Small LOCA, BWR	
15. Snubbers		15. Snubbers
16. Instruments and Controls (I&C)	16. Instruments and Controls (I&C) a. Thermostat	16. Instruments and Controls (I&C) a. Thermostat b. Transfer switch[c] c. Bistable trip[c]
17. Switchgear/relays	17. Switchgear/relays a. Relay (load) b. Circuit breaker c. Transfer switch d. Bistable trip unit	17. Switchgear Relays a. Relay b. Circuit breaker

TABLE 2.2. (contd)

TIRGALEX Groups	4-Plant PRA Components	Components Chosen by the Expert Panel
18. Valves	18. Valves a. Air-operated valve (AOV) b. Check valve c. Hydraulic valve d. Manual valve e. Motor-operated valve (MOV) f. Safety/relief valve (S/RV)	18. Valves a. Air-operated valve b. Check valve c. Hydraulic valve d. Manual valve e. Motor operated valve f. Safety/relief valve
19. Pumps	19. Pumps a Motor driven pump b. Turbine driven pump	19. Pumps a. Motor driven pump b. Turbine driven pump
20. Motors		20. Motors (included in valves, pumps, etc.)
21. Turbines	21. Turbines	21. Turbines
22. Heat exchangers	22. Heat exchangers a. Heat exchangers b. Air conditioners	22. Heat exchangers
23. Compressor		23. Compressor(d)
24. Fans/chillers	24. Fan	24. a. Chiller b. Fan
25. Batteries	25. Batteries	25. Batteries
26. Battery chargers/inverters	26. Battery chargers/inverters a. Battery chargers b. Inverters	26. Battery chargers/inverters a. Battery chargers b. Inverters c. Rectifier(e)
27. Transformers	27. Transformers	27. Transformers
28. Fuel storage racks		28. Fuel storage racks
29. Accumulator/tanks	29. Tank	29. Tanks a. Medium pressure tank b. Atmospheric pressure tank c. High pressure tank
30. AC/DC bus	30. AC/DC buses a. AC bus b. DC bus	30. AC/DC buses a. AC bus b. DC bus
		31. Bolts

(a) Represented by service water system piping for which HPI could not keep up if break occurred.
(b) Represented by piping in letdown and reactor water cleanup (RWCU) systems, for which HPI could just keep up if small break occurred.
(c) Moved by panel from S/C-17, 4-Plant PRA.
(d) Compressors in Instrument Air System.
(e) Moved by panel from non-assigned S/C (see Table 2.1).

- S/G-6: steam generator: to permit consideration of two failure modes having significantly different risk importances and aging failure rates.

- S/G-10: cables, connectors and penetrations: to permit separate consideration of cables and connectors due to differences in their importance to safety (penetrations to be part of containment).

2.2.2 Component Scores and Ranks

The following subsections present the findings generated for the factors related to both the risk significance of aging [N, A, CDFA, $\Delta R(40)$] and to the adequacy of aging management practices in reducing the risk contribution of aging [L_{ind}, P_D, $P_{R/D}$, and $(L_{act}/L_c)^2$]. As noted in Section 2.1.2, only N, A, L_{ind}, P_D, and $P_{R/D}$ are the prioritization criteria; scores are developed for them. An integrated, final ranking of all S/Cs is then presented. An overview of the rationale for these findings is presented with the findings; details are discussed in Section 2.2.3 and in Appendix D.

2.2.2.1 Risk Significance of Aging Data

Table 2.3 presents the S/C risk importance (N) values; both the precalculated values (for the PRA-based components) and the expert panel values for all components (the PRA-based, the non-PRA-based accident prevention components, and the non-PRA-based accident mitigation components) are shown. In the column for the precalculated values, the type of component is indicated for which no values were calculated.

The panel assigned a risk importance of 1.0 to S/C-1: RPV since its failure (gross rupture) results directly in core damage; sufficient cooling could not be provided. The RPV value, being the highest for accident prevention S/Cs, became the source of the value assigned for the equivalent importance of S/C-2.a: containment (BWR) and S/C-2.b: containment (other). The basis for this panel decision was the recognition of the containment's function as a barrier to radioactive releases to the public. S/C-3: other Category I structures were also given a risk importance value of 1.0 because of their role in providing support of equipment necessary for core cooling (cable trays, piping, etc.) and, in some plants, in providing a containment function.

Most of the precalculated values (for the PRA-based components) were left unchanged. Of those that were changed, the majority were reduced, reflecting both the panel's insights gained from newer, improved PRAs and the improved designs of later plants that resulted in reduced risk importance for these components. Two components, diesel generator and battery, were increased in risk importance compared to the precalculated values. The diesel generator value was raised because of the high risk contribution of loss of offsite power in the recent PRAs; the panel elevated the value for batteries to reflect their similar risk importance to the diesels.

After the S/C-1: RPV, S/C-2a: containment, S/C-3: other Category I structures, and S/C-2b: containment (other) (all of which were ranked 5), the next

TABLE 2.3. Structure/Component Risk Importances[a]

Component	Precalculated N (C D/yr)	Panel's N (C D/yr)	Score
1. RPV	AP[b]	1.0E-0	5
2.a. Containment (BWR)	AM[c]	1.0E-0	5
3. Other concrete structures	AP	1.0E-0	5
2.b. Containment (other)	AM	1.0E-0	5
30.b. DC bus	1.1E-1	1.1E-1	4
10.a. Cables	AP	1.1E-1	4
9.a. CRDM (BWR)	AP	1.0E-1	4
12. RX internals	AP	1.0E-1	4
17.b. Breaker	7.2E-2	7.2E-2	4
17.a. Relay	4.8E-2	4.8E-2	4
30.a. AC bus	4.3E-2	4.3E-2	4
29.b. Tank (atmos. pres.)	AP	2.5E-2	3
18.e. Motor operated valve	2.2E-2	2.2E-2	3
11. Diesel	9.2E-4	2.0E-2	3
10.b. Connectors	AP	2.0E-2	3
25. Battery	9.2E-4	2.0E-2	3
27. Transformer	1.2E-2	1.2E-2	3
19.b. Turbine pump	9.3E-3	9.3E-3	3
19.a. Motor pump	6.7E-3	6.7E-3	3
5.a. Large other safety pipe[d]	AP	6.4E-3	3
22. Heat exchanger	6.4E-3	6.4E-3	3
29.a. Tank (medium pres.)	AP	6.0E-3	3
16.a. Thermostat	6.0E-3	6.0E-3	3
4.b. PWR pipe (small LOCA)	AP	1.0E-3	2
5.b. Small other safety pipe[e]	AP	1.0E-3	2
4.c. BWR Pipe (small LOCA)	AP	1.0E-3	2
9.b. CRDM (PWR)	AP	1.0E-3	2
18.b. Check valve	1.8E-2	8.0E-4	2
24.a. Chillers	AP	6.0E-4	2
24.b. Fan	7.6E-3	6.0E-4	2
23. Compressor (instr. air)	AP	5.0E-4	2
18.a. Air operated valve	3.2E-2	3.2E-4	2
6.a. S/G tube	AP	3.0E-4	1
26.a. Battery charger	1.1E-4	1.1E-4	1
8. Pressurizer	AP	1.0E-4	1
7. RC P Casing	AP	1.0E-4	1
29.c. Tank (high pres.)	AP	1.0E-4	1
4.a. RC P & SE large (LOCA)	AP	1.0E-4	1
31. Bolts	AP	1.0E-4	1
18.f. Safety/relief valve	3.5E-4	1.0E-4	1
16.c. Bistable	1.2E-5	1.2E-5	1
18.c. Hydraulic valve	3.3E-4	1.0E-5	1
18.d. Manual valve	3.3E-2	1.0E-5	1
6.b. S/G shell	AP	1.0E-5	1
16.b. Transfer switch	4.7E-6	4.7E-6	1
26.b. Inverter	4.2E-4	4.7E-6	1
26.c. Rectifier	4.7E-4	4.7E-6	1
15. Snubbers	AP	1.1E-6	1
21. Turbine	1.0E-4	1.0E-6	1
28. Fuel rack	AP*[f]	1.0E-6	1
13. RPV support	AP	1.0E-7	1

(a) Caution: only the relative component position (score) is important; do not base any subsequent studies on the numerical values presented here.
(b) AP signifies a non-PRA-based accident prevention component.
(c) AM signifies a non-PRA-based accident mitigation component.
(d) 10-24 in. pipe represented by service water piping (see Table 2.2).
(e) 6-10 in. pipe represented by letdown and RWCV systems (see Table 2.2).
(f) AP* signifies the special case of fuel damage prevention.

highest ranked components (4) consisted primarily of electrical components; the numerous and widely distributed electricals, such as cables, breakers, relays, and connectors, were well represented.

Table 2.4 contains the S/C aging failure rate (A) values. As with Table 2.3, both precalculated and expert panel values are indicated, and the type of non-PRA-based component is also shown. The panel focused consideration of A on the risk significant failure mode that was used to define the component's risk importance (N).

Most of the precalculated values for the increase-in-component failure rates due to aging were left unchanged by the panel. S/C-25: battery was increased by a factor of ten and received a rank of 4; S/C-30 b: dc bus was increased by a factor of ten and S/C-30 a: ac bus was increased by two orders of magnitude, but they both were ranked only at the top of 1. The aging rates for several components were reduced compared to the precalculated values; the greatest reduction was two orders of magnitude for S/C-18.b: check valves, for which two failure mechanisms having differing rates (one high, one low) were noted.

Many of the changed aging failure rate values, as well as those for other components, resulted from the panel's consideration of the numbers of failures likely in 40 years (or component average lifetime, if less than 40 years), the doubling rates to achieve these, and then, the back-calculated aging failure rates. Other component aging failure rates were the result of the panel's relative positioning (ranking), from comparing one against another; and other values resulted from a combination of both methods. Where the panel's disaggregation of component groups did not provide sufficient distinction of the risk significant aging and failure mechanisms [as in the pitting corrosion of medium- and atmospheric-pressure tanks (S/C-29:a and b) compared to the rupture of high-pressure tanks (S/C-29c)], the panel tried to develop effective values for the competing mechanisms. This probably led component aging in the harsher environment to bias the aging failure rate (or its relative ranking).

The RPV calculations started with a low known (accepted), nonaging failure rate. This was then increased by specifying an aging failure rate that, at the end of 40 years, would represent a credible increase in vessel failure rate. This value was used for similar types of components at similar pressure and temperature, S/C-29.c: high pressure tanks, S/C-6.b: steam generator shell, and S/C-8: pressurizer. All had low aging failure rates. S/C-2.b: concrete containment and S/C-3: other Category I structures had, in the panel's view, the lowest rates (e.g., positions in the table); and as rates of the other components were generated, the numerical values of these structures resulted.

The reader is cautioned, as noted in Section 2.1.2.3, that it is the relative positioning of the components, not the absolute numerical values for the N and A factors, that is important. Many of the values, particularly for A, were the result, not the cause, of such positioning.

TABLE 2.4. Structure/Component Aging Failure Rates[a]

Component	Precalculated A ($hr^{-1}yr^{-1}$)	Panel's A ($hr^{-1}yr^{-1}$)	Score
15. Snubbers	AP[b]	5.1E-6	5
6.a. S/G tube	AP	5.0E-6	5
26.b. Inverter	4.9E-6	4.9E-6	5
11. Diesel	1.6E-6	3.6E-6	5
18.a. Motor operated valve	2.6E-7	3.6E-6	5
19.b. Turbine pump	2.7E-6	2.7E-6	5
24.a. Chillers	AP	1.5E-6	4
18.f. Safety/relief valve	6.7E-7	6.7E-7	4
31. Bolts	AP	5.1E-7	4
23. Compressor (istr. air)	AP	5.0E-7	4
18.a. Air operated valve	4.0E-7	4.0E-7	4
25. Battery	3.4E-8	3.4E-7	4
5.b. Small other safety pipe[c]	AP	3.0E-7	4
17.a. Relay	9.1E-8	2.5E-7	4
16.b. Transfer switch	2.3E-7	2.3E-7	4
19.a. Motor pump	2.2E-7	2.2E-7	4
24.b. Fan	2.1E-7	2.1E-7	4
16.a. Thermostat	1.5E-7	1.5E-7	3
16.c. Bistable	1.4E-7	1.4E-7	3
18.c. Hydraulic valve	6.5E-6	1.3E-7	3
2.a. Containment (BWR)	AM[d]	1.0E-7	3
21. Turbine	3.7E-6	1.0E-7	3
26.c. Rectifier	8.7E-8	8.7E-8	3
26.a. Battery charger	3.5E-8	3.5E-8	3
4.c. BWR Pipe (small LOCA)	AP	3.0E-8	3
10.B. Connectors	AP	2.7E-8	3
17.b. Breaker	1.6E-8	1.6E-8	2
22. Heat exchanger	1.4E-7	1.4E-8	2
18.b. Check valve	3.8E-7	3.8E-9	2
9.a. CRDM (BWR)	AP	3.0E-9	2
5.a. Large other safety pipe[e]	AP	3.0E-9	2
10.a. Cables	AP	2.7E-9	2
18.d. Manual valve	2.2E-8	2.2E-9	2
12. RX internals	AP	2.0E-9	2
27. Transformer	1.7E-8	1.7E-9	2
30.b. DC bus	1.1E-10	1.1E-9	1
30.a. AC bus	1.1E-11	1.1E-9	1
9.b. CRDM (PWR)	AP	3.0E-11	1
4.b. PWR pipe (small LOCA)	AP	1.0E-11	1
28. Fuel rack	AP*[f]	5.0E-12	1
1. RPV	AP	2.0E-12	1
29.b. Tank (atmos. pres.)	AP	1.0E-12	1
29.c. Tank (high pres.)	AP	1.0E-12	1
6.b. S/G shell	AP	1.0E-12	1
8. Pressurizer	AP	1.0E-12	1
29.a. Tank (medium pres.)	AP	1.0E-12	1
7. RC P casing	AP	1.0E-12	1
4.a. RC P & SE large (LOCA)	AP	1.0E-12	1
13. RPV support	AP	1.0E-12	1
3. Other concrete structures	AP	1.0E-13	1
2.b. Containment (other)	AM	1.0E-13	1

(a) Caution: only the relative component position (score) is important; do not base any subsequent studies on the numerical values presented here.
(b) AP signifies a non-PRA-based accident prevention component.
(c) 6-10 in. pipe represented by letdown and RWCV systems (see Table 2.2).
(d) AM signifies a non-PRA-based accident mitigation component.
(e) 10-24 in. pipe represented by service water piping (see Table 2.2).
(f) AP* signifies the special case of fuel damage prevention.

Table 2.5 presents the CDFA and ΔR (40) values for the panel's S/Cs calculated from the panel's values for N and A in Tables 2.3 and 2.4, respectively. The S/Cs are arranged in decreasing order of $\Delta R(40)$, the increase in risk in 40 years; an interim rank (e.g., without consideration of current industry aging management practices) is also calculated. A screening of S/Cs was performed by the panel based on the value selected to represent an acceptable risk contribution due to component aging, ΔR_c < 1E-7 CD/yr [see Equation (2.3)]. The S/Cs with $\Delta R(40)$ values < 1E-7 were eliminated from subsequent consideration of the adequacy of aging management practices. On this basis, the following S/Cs that were eliminated:

S/C-4a: reactor coolant piping/safe ends (large LOCA)
S/C-4b: PWR pipe (small LOCA)
S/C-6b: S/G shell
S/C-7: RCP casing
S/C-8: pressurizer
S/C-13: RPV support (PWR)
S/C-28: fuel storage rack
S/C-29a: tank (medium pressure)
S/C-29c: tank (high pressure)

The remaining S/Cs are those with risk significance of aging sufficiently high that adequate aging management practices are required to bring their risk contribution down to ΔR_c.

2.2.2.2 Adequacy of Current Industry Aging Management Practices

The S/Cs in Table 2.5 that remained after the screening (e.g., those having a ΔR > 1E-7 CD/yr in 40 years) were evaluated by the panel for adequacy of industry management practices in reducing the risk increase from component aging to the control value of < 1E-7 CD/yr.

Table 2.6 presents the data generated for the assessment of the adequacy of current industry aging management practices. The panel provided inputs for L_{ind}, P_D, and $P_{R/D}$ [see Equation (2.10)]. Values for L_c, L_{act}, and $(L_{act}/L_c)^2$ were calculated after the workshop from the panel's inputs and Equations (2.4) and (2.10), respectively. Since ΔR is directly related to the adequacy factor, $(L_{act}/L_c)^2$, [see Equation (2.7)], the greater the inadequacy of industry practices to mitigate aging $[(L_{act}/L_c)^2>1]$ the larger the risk increase. Therefore, the S/Cs are arranged with those having the highest $(L_{act}/L_c)^2$ values at the top. The table shows that very few S/Cs (of those S/Cs remaining after the screening) have adequate aging management practices [e.g., $(L_{act}/L_c)^2$ < 1.0].

Values are also shown in the table for L_c [from ΔR_c < 1E-7 (CD/yr) and Equation (2.5)], L_{ind}, P_D, $P_{R/D}$ (generated by the expert panel), and L_{act} [derived from Equation (2.10)]. As noted in Section 2.1.2, of the factors in Table 2.6, only L_{ind}, P_D, and $P_{R/D}$ are used as S/C prioritization criteria. A discussion of each of the factors in Table 2.6 is presented in the following subsections.

TABLE 2.5. Structure/Component CDFAs and 40-Year Risk Increases[a]

Component	N (C D/yr)	A ($hr^{-1} yr^{-1}$)	C DFA ($C D/yr^3$)[b]	ΔR(40) (C D/yr)	Interim Rank
2.a. Containment (BWR)	1.0E-0	1.0E-7	8.8E-4	7.0E-1	5
18.e. Motor operated valve	2.2E-2	3.6E-6	6.9E-4	5.6E-1	5
11. Diesel	2.0E-2	3.6E-6	6.3E-4	5.0E-1	5
19.b. Turbine pump	9.3E-3	2.7E-6	2.2E-4	1.8E-1	5
17.a. Relay	4.8E-2	2.5E-7	1.1E-4	8.4E-2	4
25. Battery	2.0E-2	3.4E-7	6.0E-5	4.8E-2	4
6.a. S/G tube	3.0E-4	5.0E-6	1.3E-5	1.1E-2	4
19.a. Motor pump	6.7E-3	2.2E-7	1.3E-5	1.0E-2	4
17.b. Breaker	7.2E-2	1.6E-8	1.0E-5	8.1E-3	3
24.a. Chillers	6.0E-4	1.5E-6	7.9E-6	6.3E-3	3
16.a. Thermostat	6.0E-3	1.5E-7	7.9E-6	6.3E-3	3
10.b. Connectors	2.0E-2	2.7E-8	4.7E-6	3.8E-3	3
5.b. Small other safety pipe[c]	1.0E-3	3.0E-7	2.6E-6	2.1E-3	3
9.a. CRDM (BWR)	1.0E-1	3.0E-9	2.6E-6	2.1E-3	3
10.a. Cables	1.1E-1	2.7E-9	2.6E-6	2.1E-3	3
23. Compressor (instr. air)	5.0E-4	5.0E-7	2.2E-6	1.8E-3	3
12. RX internals	1.0E-1	2.0E-9	1.8E-6	1.4E-3	3
18.a. Air operated valve	3.2E-4	4.0E-7	1.1E-6	9.0E-4	2
24.b. Fan	6.0E-4	2.1E-7	1.1E-6	8.8E-4	2
30.b. DC bus	1.1E-1	1.1E-9	1.1E-6	8.5E-4	2
22. Heat exchanger	6.4E-3	1.4E-8	7.8E-7	6.3E-4	2
18.f. Safety/relief valve	1.0E-4	6.7E-7	5.9E-7	4.7E-4	2
31. Bolts	1.0E-4	5.1E-7	4.5E-7	3.6E-4	2
30.a. AC bus	4.3E-2	1.1E-9	4.1E-7	3.3E-4	2
4.c. BWR pipe (small LOCA)	1.0E-3	3.0E-8	2.6E-7	2.1E-4	2
26.b. Inverter	4.7E-6	4.9E-6	2.0E-7	1.6E-4	2
27. Transformer	1.2E-2	1.7E-9	1.8E-7	1.4E-4	2
5.a. Large other safety pipe[d]	6.4E-3	3.0E-9	1.7E-7	1.3E-4	2
15. Snubbers	1.1E-6	5.1E-6	4.9E-8	3.9E-5	1
26.a. Battery charger	1.1E-4	3.5E-8	3.4E-8	2.7E-5	1
18.b. Check valve	8.0E-4	3.8E-9	2.7E-8	2.1E-5	1
1. RPV	1.0E-0	2.0E-12	1.8E-8	1.4E-5	1
16.c. Bistable	1.2E-5	1.4E-7	1.5E-8	1.2E-5	1
18.c. Hydraulic valve	1.0E-5	1.3E-7	1.1E-8	9.1E-6	1
16.b. Transfer switch	4.7E-6	2.3E-7	9.5E-9	7.6E-6	1
26.c. Rectifier	4.7E-6	8.7E-8	3.6E-9	2.9E-6	1
21. Turbine	1.0E-6	1.0E-7	8.8E-10	7.0E-7	1
3. Other concrete structures	1.0E-0	1.0E-13	8.8E-10	7.0E-7	1
2.b. Containment (other)	1.0E-0	1.0E-13	8.8E-10	7.0E-7	1
9.b. CRDM (PWR)	1.0E-3	3.0E-11	2.6E-10	2.1E-7	1
29.b. Tank (atmos. pres.)	2.5E-2	1.0E-12	2.2E-10	1.8E-7	1
18.d. Manual valve	1.0E-5	2.2E-9	1.9E-10	1.5E-7	1
					[e]
4.b. PWR Pipe (small LOCA)	1.0E-3	1.0E-11	8.8E-11	7.0E-8	1
29.a. Tank (medium pres.)	6.0E-3	1.0E-12	5.3E-11	4.2E-8	1
29.c. Tank (high pres.)	1.0E-4	1.0E-12	8.8E-13	7.0E-10	1
4.a. RC P & SE large (LOCA)	1.0E-4	1.0E-12	8.8E-13	7.0E-10	1
7. RC P casing	1.0E-4	1.0E-12	8.8E-13	7.0E-10	1
8. Pressurizer	1.0E-4	1.0E-12	8.8E-13	7.0E-10	1
6.b. S/G shell	1.0E-5	1.0E-12	8.8E-14	7.0E-11	1
28. Fuel rack	1.0E-6	5.0E-12	4.4E-14	3.5E-11	1
13. RVP support	1.0E-7	1.0E-12	8.8E-16	8.0E-13	1

(a) Caution: only the relative component position (rank) is important; do not base any subsequent studies on the numerical values presented here.
(b) CDFA (CD/yr) = N (C D/yr) x A (1/hr yr) x 8760 (hr/yr).
(c) 6-10 in. pipe represented by letdown and RWCV systems (see Table 2.2).
(d) 10-24 in. pipe represented by service water piping (see Table 2.2).
(e) Components below this line were eliminated from further consideration by screening criterion (ΔR_c < 1E-7 C D/yr).

TABLE 2.6. Adequacy of Current Industry Aging Management Practices[a] (Truncated to 40 years)

Component	L_c (mo)	L_{ind} (mo)	P_D	$P_{R/D}$	L_{act} (mo)	$(L_{act}/L_c)^2$	Interim Rank
2.a. Containment (BWR)	0.1	18.0	0.90	0.80	25.0	62500.00	5
10.a. Cables	3.3	60.0	0.10	0.90	480.0[b]	21155.70	5
5.b. Small other safety pipe[c]	3.3	60.0	0.10	0.90	480.0[b]	21155.70	5
10.b. Connectors	2.5	60.0	0.20	0.90	333.3	17774.22	5
6.a. S/G tube	1.5	36.0	0.50	0.50	144.0	9216.00	4
19.b. Turbine pump	0.4	12.0	0.50	0.90	26.7	4455.56	4
17.a. Relay	0.5	6.0	0.20	0.90	33.3	4435.56	4
11. Diesel	0.2	3.0	0.30	0.90	11.1	3080.25	4
5.a. Large other safety pipe[d]	4.1	18.0	0.10	0.90	200.0	2379.49	4
12. RX internals	4.1	18.0	0.10	0.90	200.0	2379.49	4
18.e. Motor operated valve	0.2	3.0	0.70	0.90	4.8	576.00	3
17.b. Breaker	1.7	18.0	0.50	0.90	40.0	553.66	3
4.e. BWR pipe (small LOCA)	10.5	36.0	0.20	0.90	200.0	362.90	3
19.a. Motor pump	1.5	12.0	0.50	0.90	26.7	316.84	3
24.a. Chillers	1.9	18.0	0.70	0.90	28.6	226.50	3
16.a. Thermostat	1.9	18.0	0.70	0.90	28.6	226.50	3
1. RPV	40.5	120.0	0.90	0.10	480.0[b]	140.42	3
25. Battery	0.7	6.0	0.90	0.90	7.4	111.72	3
23. Compressor (instr. air)	3.6	6.0	0.20	0.90	33.3	85.56	2
3. Other concrete structures	57.3	60.0	0.10	0.50	480.0[b]	70.22	2
2.b. Containment (other)	57.3	60.0	0.10	0.50	480.0[b]	70.22	2
18.a. Air operated valve	5.1	18.0	0.50	0.90	40.0	61.47	2
30.b. DC bus	5.2	18.0	0.50	0.90	40.0	59.14	2
9.a. CRDM (BWR)	3.3	18.0	0.90	0.90	22.2	45.29	2
18.b. Check valve	32.9	18.0	0.10	0.90	200.0	36.97	2
24.b. Fan	5.1	18.0	0.70	0.90	28.6	31.47	2
22. Heat exchanger	6.1	3.0	0.10	0.90	33.3	29.81	2
31. Bolts	8.0	18.0	0.50	0.90	40.0	25.00	2
30.a. AC bus	8.3	18.0	0.50	0.90	40.0	23.23	2
18.f. Safety/relief valve	7.0	18.0	0.90	0.90	22.2	10.05	2
27. Transformer	12.7	18.0	0.70	0.90	28.6	5.06	1
26.b. Inverter	11.9	12.0	0.50	0.90	26.7	5.02	1
16.b. Transfer switch	55.1	18.0	0.20	0.90	100.0	3.28	1
15. Snubbers	24.2	18.0	0.90	0.90	22.2	0.85	1
18.c. Hydraulic valve	50.3	18.0	0.50	0.90	40.0	0.64	1
21. Turbine	181.3	60.0	0.90	0.50	133.3	0.55	1
16.c. Bistable	44.2	18.0	0.70	0.90	28.6	0.42	1
18.d. Manual valve	386.6	60.0	0.50	0.60	200.0	0.27	1
26.a. Battery charger	29.2	12.0	0.90	0.90	14.8	0.26	1
29.b. Tank (atmos. pres.)	362.6	12.0	0.10	0.70	171.4	0.22	1
26.c. Rectifier	89.7	12.0	0.50	0.90	26.7	0.09	1
29.a. Tank (medium pres.)	740.2	12.0	0.10	0.70	171.4	0.05	1
9.b. CRDM (PWR)	331.0	18.0	0.50	0.90	40.0	0.01	1

(a) *Caution: only the relative component position (rank) is important; do not base any subsequent studies on the numerical values presented here.*
(b) L_{act} for these components was truncated to a maximum of 40 years (480 mo).
(c) 6-10 in. pipe represented by letdown and RWCV systems (see Table 2.2).
(d) 10-24 in. pipe represented by service water piping (see Table 2.2).

The Current Industry Surveillance/Test Factor (L_{ind})

In generating the values for L_{ind}, the panel considered the surveillance/test interval representative of current industry practice that was relevant to the risk-significant failure mode (the one used to generate N). The panel concluded that for many mechanical and electrical components, the current surveillance/operability tests required by technical specifications were of limited value in identifying these failure modes. Often, only those tests conducted during the refueling outages, in conjunction with preventive maintenance activities, were considered to be relevant. This was the rationale for the many 18-month assignments made for L_{ind} that can be observed in Table 2.6.

In assessing how frequently a surveillance/test relevant to a failure mode was likely to be made, a composite value often had to be determined. Some features of the failure mode might be observed more frequently, while others might be observed less frequently. An example is the diesel generator, where some panelists felt that relevant observations could be made as frequently as once a shift, but where others felt that only during the 18-month test/maintenance activity would information specific to failure mode be generated; a 3-month L_{ind} was finally agreed upon. These L_{ind} assessments were very difficult for the panel. The L_{ind} value for RPV became the longest, and was based upon the 10-year inspection requirement; those for containment and concrete structures were arrived at by consensus on the time span within which physical evidence of a problem (spalling, liner bulging, etc.) could be expected to be observable.

Only one component (S/C-22: heat exchanger) has an L_{ind} less than its L_c (3.0 versus 6.1 mo). However, even this interval is too long. From Equation (2.14) and the values for P_D and L_c in Table 2.6, an acceptable industry surveillance/test interval [e.g., one required to control risk to a desirable level $(L_{ind})_{acc}$] should not exceed 0.3 months. This is due to the low P_D; L_c, being >1, is more than adequate. When, as is the case for many of the components, L_{ind} is larger than L_c and P_D is very low, then the required reduction in the interval is even greater. For example, S/C-10: cables would require an $(L_{ind})_{acc}$ of 0.33 months compared to the value for L_{ind} of 60 months.

Note that even where P_D and $P_{R/D}$ are each ~1.0, if L_{ind} is $>L_c$, ΔR will increase. S/C-25: battery, S/C-9a: CRDM (BWR) and S/C-2a: containment (BWR) are examples of this. Therefore, an adequate frequency of surveillances/tests that are relevant to the risk significant failure mode is necessary for acceptable management of component aging.

The Probability of Successful Aging Detection Factor (P_D)

After determining the value for L_{ind}, the panel turned to an assessment of how successful the surveillance/test chosen as relevant to detecting the risk importance failure mode of the component would be in detecting the aging mechanisms responsible for that failure. (We assume, here, only aging-caused failures, using the definition of aging given in Section 1.2.)

The surveillance/test methods for many components were judged to be ineffective in detecting aging; these were given P_D values below 0.5, many in the 0.1 - 0.3 range.

The present surveillance/test methods were judged to be truly effective in detecting aging mechanisms for less than one-fourth of the components. These were given a P_D value of 0.9. Values larger than 0.9 were not used in order to indicate that similar components might not be inspected for the aging mechanism. Values less than 0.1 were not used because of the overwhelming effect this would have on the total risk contribution value. (Moreover, the value 0.1 represents the probability that over a 40-year period, at least once in 48 months detection would be successful. For those components that are completely overhauled or replaced, this is reasonable. For major structures such as piping, the P_D and L_{ind} values generated by the panel suggest that this may not be conservative.)

The impact of a low P_D value is that more frequent surveillances/tests are required to ensure the detection of aging degradation within the interval L_c. We noted this effect when we discussed the L_{ind} values for heat exchangers and cables in the preceding subsection.

The Probability of Successful Aging Mitigation Factor ($P_{R/D}$)

The panel next determined the probability of successfully mitigating the risk significant aging mechanism once it was detected (e.g., to turn the aging clock back to zero or as-good-as-new). Most components were given a 0.9 value. Values larger than 0.9 were not used by the panel in order to indicate that 1) repairs appearing to be successful may not always be successful, or 2) full operational performance might be achieved but only with an as-good-as-old condition (e.g., the aging clock is not fully turned back to zero).

A few components were in the 0.5 range and RPV received a 0.1. A value of 0.5 was given for one of several reasons; the following are examples:

- The observed aging-caused defect may not be considered to be risk significant, so it would be unlikely that a repair to as-good-as-new conditions would be made (S/C-2b: containment-other).
- Only the observable defect would be repaired; the age of the rest of the component would not be returned to zero (S/C-6.a: steam generator tube).

The $P_{R/D}$ value of 0.1 for the RPV was based on the lack of evidence confirming the benefits of vessel annealing. Also, recent laboratory data indicate that irradiation after annealing may result in a faster rate of embrittlement. Mitigation of aging via an NRC-mandated plant shutdown, should aging exceed the screening criterion (RG-1.99 Rev. 2), was not considered relevant to the intent of $P_{R/D}$, namely, to return the aging clock to zero and return the component to full functionality (aging, in this case, would cease, but the component would no longer be functional).

The Factors Relating to Adequacy of Industry Aging Management Practices (L_{act}, L_c, and L_{act}/L_c)

L_{act}, the actual (effective) interval representative of current industry practices, was calculated directly from L_{ind}, P_D, and $P_{R/D}$ subsequent to the workshop. Some very high values were obtained. To stay within the guidelines of considering aging only in the context of the current 40-year license period, the calculations of L_{act} were truncated to 480 months. As observed in Table 2.5, many components had effective intervals in the hundreds of months, including five components with intervals of 480 months.

A critical parameter in comparing aging management practices across components is $(L_{act}/L_c)^2$. This parameter is directly proportioned to $\Delta R/\Delta R_c$; hence, it is a ratio index of the aging risk under present aging management practices to the acceptable level [see Equations (2.7) through (2.9)]. For aging management practices to be acceptable, $(L_{act}/L_c)^2 < 1$. Table 2.6 shows that many components, even those having an L_{act} that might be considered to be low (such as S/C-25: battery, with an L_{act} of only 7.4 mo.), have $\Delta R > \Delta R_c$. In the case of the battery, $(L_{act}/L_c)^2$ is 112; for S/C-2a: containment (BWR-MK-I), it is 6.2E+4. The reasons for a high $(L_{act}/L_c)^2$ can be three-fold:

1. A highly aging risk significant component (CDFA is high) will necessitate a very low L_c in order to keep ΔR low [see Equation (2.5)]. This is the case for S/C-2a:containment (BWR-MK-I); its L_c is only 0.1 month.

2. A component with a long L_{ind} and/or with a low P_D or $P_{R/D}$ will have a long L_{act}. This is the case for S/C-3:other Category I structures; its L_{act} is 480.

3. Combinations of the above 1. and 2.

Whatever the reason, a high $(L_{act}/L_c)^2$ indicates the need for improved aging management practices.

Conversely, relatively poor aging management practices may not lead to an increase in a component's risk contribution. This is the case when the component's L_c is high (due to a low risk significance of aging, CDFA). Such is the case for S/C-18d: manual valve, with an $(L_{act}/L_c)^2$ of only 0.27 even though its L_{act} is 200, due to a P_D of only 0.5; its L_c is very long (386.6 mo.).

2.2.3 Final Integrated Ranking of the Structures and Components

Table 2.7 presents the values for the factors used in the final RSCAAMP calculations [Equation (2.15)] and in the final ranking of the S/Cs. The S/Cs are arranged in decreasing order of contribution to plant risk increase. Those components with $\Delta R(40)$ values were below 1E-7 CD/yr and were not evaluated for adequacy of aging management practice (as identified in Section 2.2.2) are included in this table; their rankings are placed at the nominal value of one (1).

TABLE 2.7. Factor Values and Final Ranking of Structures/Components[a] (Truncated to 40 years)

Component	N (C D/yr)	A ($hr^{-1} yr^{-1}$)	L_{ind} (mo.)	P_D	$P_{R/D}$	Risk Increase ΔR (C D/yr)[b]	Final Rank
5.b. Small other safety pipe[c]	1.0E-3	3.0E-7	60.0	0.10	0.90	2.1E-3[d]	5
10.a. Cables	1.1E-1	2.7E-9	60.0	0.10	0.90	2.1E-3[d]	5
2.a. Containment (BWR)	1.0E-0	1.0E-7	18.0	0.90	0.80	1.9E-3	5
10.b. Connectors	2.0E-2	2.7E-8	60.0	0.20	0.90	1.8E-3	5
6.a. S/G tube	3.0E-4	5.0E-6	36.0	0.50	0.50	9.5E-4	5
19.b. Turbine pump	9.3E-3	2.7E-6	12.0	0.50	0.90	5.4E-4	4
17.a. Relay	4.8E-2	2.5E-7	6.0	0.20	0.90	4.0E-4	4
11. Diesel	2.0E-2	3.6E-6	3.0	0.30	0.90	2.7E-4	4
12. RX internals	1.0E-1	2.0E-9	18.0	0.10	0.90	2.4E-4	4
17.b. Breaker	7.2E-2	1.6E-8	18.0	0.50	0.90	5.6E-5	3
18.e. Motor operated valve	2.2E-2	3.6E-6	3.0	0.70	0.90	5.6E-5	3
4.c. BWR pipe (small LOCA)	1.0E-3	3.0E-8	36.0	0.20	0.90	3.7E-5	3
19.a. Motor pump	6.7E-3	2.2E-7	12.0	0.50	0.90	3.2E-5	3
5.a. Large other safety pipe[e]	6.4E-3	3.0E-9	18.0	0.10	0.90	2.3E-5	3
16.a. Thermostat	6.0E-3	1.5E-7	18.0	0.70	0.90	2.2E-5	3
24.a. Chillers	6.0E-4	1.5E-6	18.0	0.70	0.90	2.2E-5	3
1. RPV	1.0E-0	2.0E-12	120.0	0.90	0.10	1.4E-5[d]	3
25. Battery	2.0E-2	3.4E-7	6.0	0.90	0.90	1.1E-5	3
23. Compressor (instr. air)	5.0E-4	5.0E-7	6.0	0.20	0.90	8.4E-6	3
18.a. Air operated valve	3.2E-4	4.0E-7	18.0	0.50	0.90	6.2E-6	2
30.b. DC bus	1.1E-1	1.1E-9	18.0	0.50	0.90	5.9E-6	2
9.a. CRDM (BWR)	1.0E-1	3.0E-9	18.0	0.90	0.90	4.5E-6	2
18.b. Check valve	8.0E-4	3.8E-9	18.0	0.10	0.90	3.7E-6	2
24.b. Fan	6.0E-4	2.1E-7	18.0	0.70	0.90	3.1E-6	2
22. Heat exchanger	6.4E-3	1.4E-8	3.0	0.10	0.90	3.0E-6	2
31. Bolts	1.0E-4	5.1E-7	18.0	0.50	0.90	2.5E-6	2
30.a. AC bus	4.3E-2	1.1E-9	18.0	0.50	0.90	2.3E-6	2
18.f. Safety/relief valve	1.0E-4	6.7E-7	18.0	0.90	0.90	1.0E-6	2
2.b. Containment (other)	1.0E-0	1.0E-13	60.0	0.10	0.50	7.0E-7[d]	2
3. Other concrete structures	1.0E-0	1.0E-13	60.0	0.10	0.50	7.0E-7[d]	2
27. Transformer	1.2E-2	1.7E-9	18.0	0.70	0.90	5.1E-7	1
26.b. Inverter	4.7E-6	4.9E-6	12.0	0.50	0.90	5.0E-7	1
16.b. Transfer switch	4.7E-6	2.3E-7	18.0	0.20	0.90	3.3E-7	1
15 Snubbers	1.1E-6	5.1E-6	18.0	0.90	0.90	8.4E-8	1
18.c. Hydraulic valve	1.0E-5	1.3E-7	18.0	0.50	0.90	6.3E-8	1
21. Turbine	1.0E-6	1.0E-7	60.0	0.90	0.50	5.4E-8	1
16.c. Bistable	1.2E-5	1.4E-7	18.0	0.70	0.90	4.2E-8	1
18.d. Manual valve	1.0E-5	2.2E-9	60.0	0.50	0.60	2.7E-8	1
26.a. Battery charger	1.1E-4	3.5E-8	12.0	0.90	0.90	2.6E-8	1
29.b. Tank (atmos. pres.)	2.5E-2	2.0E-12	12.0	0.10	0.70	2.2E-8	1
26.c. Rectifier	4.7E-6	8.7E-8	12.0	0.50	0.90	8.9E-9	1
29.a. Tank (medium pres.)	6.0E-3	1.0E-12	12.0	0.10	0.70	5.4E-9	1
9.b. CRDM (PWR)	1.0E-3	3.0E-11	18.0	0.50	0.90	1.5E-9	1
							(f)
8. Pressurizer	1.0E-4	1.0E-12	18.0	0.90	0.90	1.2E-12	1
6.b. S/G shell	1.0E-5	1.0E-12	--	--	--	--	1
29.c. Tank (high pres.)	1.0E-4	1.0E-12	--	--	--	--	1
4.a. RC P & SE large (LOCA)	1.0E-4	1.0E-12	--	--	--	--	1
7. RC P casing	1.0E-4	1.0E-12	--	--	--	--	1
13. RPV support	1.0E-7	1.0E-12	--	--	--	--	1
28. Fuel rack	1.0E-6	5.0E-12	--	--	--	--	1
4.b. PWR pipe (small LOCA)	1.0E-3	1.0E-12	--	--	--	--	1

(a) Caution: only the relative component position (rank) is important; do not base any subsequent studies on the numerical values presented here.
(b) To obtain the units for ΔR, (C D/yr), A is multiplied by 8760 hrs yr^{-1} and L_{ind} is multiplied by 12 yr^{-1} (both P_D and $P_{R/D}$ are probabilities).
(c) 6-10 in. pipe represented by letdown and RWCV systems (see Table 2.2).
(d) L_{act} for these components was truncated to a maximum of 40 years (480 months).
(e) 10-24 in. pipe represented by service water piping (see Table 2.2).
(f) Components below this line were eliminated from further consideration by screening criterion (ΔR_c < 1E-7 C D/yr); the pressurizer was not eliminated until a post-workshop re-analyses.

In Table 2.7, the risk increases and the final S/C rankings are based upon the truncation of L_{act} to 480 months (40 years), as explained in Section 2.2.2. Even with truncation, the effect on component aging risk contribution of the adequacy of industry aging management practices is marked. This can be seen by comparing the relative positions of S/C-5b: small other safety related piping and S/C-2a: containment (BWR-MK-I) before and after L_{act} is factored into the component risk contribution equation. Table 2.5 shows that S/C-5b is lower than S/C-2a by a factor of 300 in risk increase due to aging $\Delta R(40)$, and is ranked 3 compared to S/C-2a which is ranked 5. However, Table 2.6 shows that L_{act} for S/C-5b is almost a factor of 20 larger than L_{act} for S/C-2a (e.g., its aging management is that much less effective). Due to L_{act} entering the RSCAAMP equation as the square, the net result is that S/C-5b has a greater risk contribution than does S/C-2a, and is now at the top of the rankings.

If L_{act} had not been truncated, several components would have L_{act} values well above 480 months and, as a result, would have higher final risk increase values. This impacts the consideration of plant relicensing. If a nuclear plant were allowed to operate beyond 40 years, the ineffectiveness of aging management practices represented by the components' untruncated values would result in continuing and increasing risk. (Note: this assumes no change in those practices.) Of particular concern would be S/C-1: RPV, S/C-2b: containment-other, and S/C-3: other concrete structures since these are permanent structures (though the possibility of RPV replacement has been considered). Untruncated L_{act} values increase from 480 months to 1333 months for the RPV, and to 1200 months each for the other two structures. The final ΔR value for the RPV increased by one order of magnitude; for the other structures, ΔR increased by one and one-half orders of magnitude.

We again caution that appropriate care must be exercised in the interpretation and use of the numerical values in this table. Only the relative ranks of the components should be used as a basis for future studies.

Table 2.8 shows the scores for the five prioritization criteria and the final rankings of the S/Cs. All components are shown in this table, including those eliminated by the screening criterion from consideration of the adequacy of aging practices.

2.2.4 Detailed Rationale for the Panel's Findings

Expert panel judgment was used to alter the precalculated values of N and A for the PRA-based components where deemed appropriate; to supply values of N and A for the non-PRA-based components; and to supply values for L_{ind}, P_D and $P_{R/D}$ for all S/Cs having risk significance of aging over 40-years, $\Delta R(40)$, greater than the screening criterion.

The rationale expressed by the panel for its decisions as recorded by the rapporteurs. The detailed rationale is presented in Appendix D. These rationale are a line-by-line itemization of the values given for the S/C listings in Table 2.7. Appendix D is arranged by S/C number (e.g., S/C.1 RPV, etc.).

TABLE 2.8. Structures/Components Prioritization Criteria Scores and Final Rankings (Truncated to 40 years)

Component	N Score	A Score	L_{ind} Score	P_D Score	$P_{R/D}$ Score	Risk Increase Rank
5.b. Small other safety pipe[a]	2	4	4	5	1	5[b]
10.a. Cables	4	2	4	5	1	5[b]
2.a. Containment (BWR)	5	3	3	1	2	5
10.b. Connectors	3	3	4	4	1	5
6.a. S/G tube	1	5	4	3	4	5
19.b. Turbine pump	3	5	2	3	1	4
17.a. Relay	4	4	1	4	1	4
11. Diesel	3	5	1	3	1	4
12. RX internals	4	2	3	5	1	4
17.b. Breaker	4	2	3	3	1	3
18.e. Motor operated valve	3	5	1	2	1	3
4.c. BWR pipe (small LOCA)	2	3	4	4	1	3
19.a. Motor pump	3	4	2	3	1	3
5.a. Large other safety pipe[c]	3	2	3	5	1	3
16.a. Thermostat	3	3	3	2	1	3
24.a. Chillers	2	4	3	2	1	3
1. RPV	5	1	5	1	5	3[b]
25. Battery	3	4	1	1	1	3
23. Compressor (instr. air)	2	4	1	4	1	3
18.a. Air operated valve	2	4	3	3	1	2
30.b. DC bus	4	1	3	3	1	2
9.a. CRDM (BWR)	4	2	3	1	1	2
18.b. Check valve	2	2	3	5	1	2
24.b. Fan	2	4	3	2	1	2
22. Heat exchanger	3	2	1	5	1	2
31. Bolts	1	4	3	3	1	2
30.a. AC bus	4	1	3	3	1	2
18.f. Safety/relief valve	1	4	3	1	1	2
2.b. Containment (other)	5	1	4	5	4	2[b]
3. Other concrete structures	5	1	4	5	4	2[b]
27. Transformer	3	2	3	2	1	1
26.b. Inverter	1	5	2	3	1	1
16.b. Transfer switch	1	4	3	4	1	1
15. Snubbers	1	5	3	1	1	1
18.c. Hydraulic valve	1	3	3	3	1	1
21. Turbine	1	3	4	1	4	1
16.c. Bistable	1	3	3	2	1	1
18.d. Manual valve	1	2	4	3	3	1
26.a. Battery charger	1	3	2	1	1	1
29.b. Tank (atmos. pres.)	3	1	2	5	2	1
26.c. Rectifier	1	3	2	3	1	1
29.a. Tank (medium pres.)	3	1	2	5	2	1
9.b. CRDM (PWR)	2	1	3	3	1	1
						[d]
8. Pressurizer	1	1	--	--	--	1
6.b. S/G shell	1	1	--	--	--	1
29.c. Tank (high pres.)	1	1	--	--	--	1
4.a. RC P & SE large (LOCA)	1	1	--	--	--	1
7. RC P casing	1	1	--	--	--	1
13. RPV support	1	1	--	--	--	1
28. Fuel rack	1	1	--	--	--	1
4.b. PWR pipe (small LOCA)	2	1	--	--	--	1

(a) 6-10 in. pipe represented by letdown and RWCV systems (see Table 2.2).
(b) L_{act} for these components was truncated to a maximum of 40 years (480 months).
(c) 10-24 in. pipe represented by service water piping (see Table 2.2).
(d) Components below this line were eliminated from further consideration by screening criterion.

2.3 TECHNICAL INSIGHTS AND DISCUSSION

Analyses of the workshop process, the application of the RSCAAMP methodology, and the panel's findings offer some additional technical insights. These are discussed in the following sections.

2.3.1 The Workshop Process

The workshop was deemed by the panel members to be well-organized and executed, and successful in meeting its objectives. The varied and relevant expertise within the panel membership led to a useful synergism; pre- and in-workshop training processes brought the panel to an early state of readiness to meet the prioritization objectives; and the formulation of the mathematically-derived RSCAAMP model provided the structure (through the five factors) needed to allow the panel's deliberations to be productive. In addition, the workshop process allowed the panel to supply relative rankings of the components within each factor, from which the corresponding estimated values were used to calculate the final rankings. All members felt that more time was needed; a post-workshop conference call and individual contacts to refine some of the data confirmed this.

2.3.2 Application of the RSCAAMP Methodology

2.3.2.1 Application to TIRGALEX S/C

The application of the RSCAAMP to the TIRGALEX list of S/Cs presented several issues that the panel had to deal with. The first issue was the need to develop methodologies to deal with the non-PRA-based components within the list. For the accident-prevention subset of these S/Cs, the panel had to develop its own set of risk importances from its intellectual perceptions rather than available hard data; out of necessity, compromises developed. For the accident-mitigation subset of S/Cs, an arbitrary decision on their equivalent importance based upon defense-in-depth considerations was used to facilitate the process.

The second issue resulting from the TIRGALEX S/C list was the need to disaggregate certain groups to more adequately consider component risk importances, aging failure rates, etc. While disaggregation was needed and very useful, considerable effort was required to achieve it; more time was needed and compromises were made. Even with disaggregation, grouping of components still occurred (e.g., S/G 18: valves were partially disaggregated to S/C-18a: motor-operated valves, but not to a single component like reactor pressure vessel). In many instances, the panel developed, in essence, a single surrogate component for the group (e.g., one having the higher risk importance or aging failure rate) to develop its N and A rankings. The summing process that the basic RSCA model uses to determine ΔR is equivalent to assuming that a high degree of correlation exists between the aging-related failures of the group. This would tend to produce a higher ΔR value than would be produced for a single component group (such as the RPV).

The most desirable situation would be that involved in performing a plant-specific RSCAAMP assessment. Specific, individual components (e.g., "check valve 5952") would be evaluated. While the RSCAAMP model is applicable to this, unless there was a plant-specific PRA and better aging failure data base available, several of the problems the panel faced would still exist. Moreover, to provide guidance to a generic NRC aging research program, an evaluation at the level of "check value 5952" is not necessary.

2.3.2.2 Application of the RSCAAMP Factors

In order to achieve its goal, the panel developed working definitions for the factors N, A, L_{ind}, P_D, $P_{R/D}$, and implemented a screening of S/Cs based upon CDFA over 40 years <1E-7 CD/yr. The working definitions focused deliberations on risk-significant failure modes (N); the aging failure rates (A), surveillance/test intervals (L_{ind}), and aging degradation mechanisms responsible for these failure modes; and the methods required to detect (P_D) and mitigate them ($P_{R/D}$). As a result of using this focused approach, it became apparent that the data bases for N and A were limited, that current requirements for inspection/tests do not adequately address the risk-significant failure mode, that successful detection of aging on one component (if that were to be achieved) doesn't guarantee that the redundant component in that system (or similar components in other systems) will be examined, and that current repair practices may not effectively reduce the risk contributions of aging (e.g., even if some S/G tubes are plugged, the aging clock for the rest of the S/G is still ticking). The screening criterion worked well since all S/Cs below 5.1E-7 were ranked 1.

2.3.3 Unexpected Findings in the Component Rankings

The component rankings produced some unexpected findings (surprises).

The panel members, on an individual basis, performed a pre-workshop homework assignment to prioritize the TIRGALEX S/Cs after reading the pre-workshop material sent to them. The composite results of the members' efforts are presented in Table 2.9. A comparison of this table to Table 2.7 shows that in doing its homework, the panel ranked too high RPV, pressurizer, snubbers, some valves, fuel storage racks, turbines, and tanks; and it ranked too low other safety-related piping (specifically the small piping), containment, cables, connectors and reactor internals.

Perhaps the most significant surprise is the number of high-ranked component groups representing components that are abundant and diversely spread throughout many plant systems (e.g., "abundant small components," as opposed to singly redundant components such as the RPV). If we include within this class of components S/C-5b: small other safety related piping, S/C-10a: cables, S/C-10b: connectors, and S/C-6a: S/G tubes, we find that in the top rank (5), four of the five component groups were of this abundant small components category. It is striking that while both piping and cables are ranked at the top, they are, in fact, not explicitly modeled in the PRAs. These components possess analogous roles; cables form the "connection" between major electrical components, while piping forms the "connection" between critical water system

TABLE 2.9. Panel's Preliminary (Homework) Ranking of Components

Rankings Between 4-5
6. Steam generator (PWR)
1. Reactor pressure vessel

Rankings Between 3-4
4. Reactor cooling piping and safe ends
8. Pressurizer (PWR)
10. Cables, connectors, and penetrations
11. Emergency diesel generator
17. Switchgear and relays
18. Valves
2. Containment (metal and concrete)
9. Control rod drive mechanism
14. Recirculation piping safe ends (BWR)
16. Instruments and controls
25. Batteries

Rankings Between 2-3
26. Battery chargers/inverters
3. Other category/concrete structures
5. Other safety-related piping
12. Reactor internals
19. Pumps
30. AC/DC buses
15. Snubbers
22. Heat exchangers
7. Reactor coolant pump casting
13. RPV Support (sliding foot) (PWR)
21. Turbines
23. Compressors
28. Fuel storage racks
29. Accumulator tanks

Rankings Between 1-2
20. Motors
24. Fans/chillers
27. Transformers

components. The piping class may be easily expanded to include S/G tubing as a specialized collection of piping; and connectors can be considered to be the linkages between runs of cables.

Conversely, PRAs have shown the failure of concrete structures during seismic events to be a very important (often dominant) contributor to risk. It is only the extremely low aging failure rate for these structures with respect to their seismic fragility which keeps their ranking low. Again, the RSCAAMP model considers more than one factor in assessing the risk contribution of a component.

That diesels did not end up in the top rank may be surprising since they are shown to be significant contributors to plant risk. This results because the aggregate of RSCAAMP factors is even more risk significant for certain other components (e.g., other small pipe, cables, etc.) than for diesels, as Table 2.7 indicates.

2.3.4 Role of the RSCAAMP Factors in the Component Rankings

The RSCAAMP factors were analyzed to understand reasons for the surprises. The interplay of the various RSCAAMP factors suggests some answers. This was alluded to above in the discussion of the low ranking of concrete structures. A comparison of the pre-workshop homework results to workshop results is also enlightening. For example, the high ranking of the RPV in the homework was probably due to its high risk importance; however, the low aging failure rate (A) and high detection capability (P_D) generated at the workshop reduce its overall risk significance compared to the intuitive position. In many cases, the higher ranking after the RSCAAMP activity, compared to the homework, resulted from the considerations of the effectiveness of industry aging management practices. The forced examination of the various risk factors in the RSCAAMP model during the workshop provided these different perspectives.

A more rigorous examination of the role that the various RSCAAMP factors have in a component's final risk contribution rank was performed. For this analysis, we addressed only the top two aging risk ranks (5 and 4) which contain the top two orders of magnitude of risk change.

The matrix in Table 2.10 displays the component groups ranked 5 or 4 in the panel's final evaluation. Beside them are three columns, one each for N, A, and $(L_{act})^2$, where $(L_{act})^2$ incorporates the industry aging management factors L_{ind}, P_D, and $P_{R/D}$. If the value of the individual factor (N, A, etc.) is within the top order of magnitude for that factor, an "X" is assigned under that column. Tabulations are then made of 1) the number of factors contributing to the component's rank (by counting horizontally across the table), and 2) the number of times a factor was the principal contributor to the top-ranked components (by counting vertically down the table).

From N-1 and N-2, we can conclude that no single factor dominated the ranking of these top-ranked components. From N-1, we see that only two groups (cables and RX internals) had as many as two factors contributing to their ranking. From N-2, we see that each factor had an approximately equal influence on component rankings; N was a principle contributor 3 times, A 3 times and $(L_{act})^2$ 4 times. The component group relays had no principle contributing factor, but all three factors were just under the top order of magnitude. Steam generator tubing is ranked high specifically because of its high aging failure rate (A); its risk importance (N) is extremely low and A acts to raise the overall rank.

While this analyses shows that, on an overall basis, each factor had an approximately equal influence on components ranked 4 and 5, there is another perspective: the importance of including component aging and its management in an assessment of plant risk. Table 2.10 illustrates the importance for the

TABLE 2.10. Analysis of the Contribution of RSCAAMP Factors to the Final Risk Contribution Rank of Top-Ranked Components

Component Category	Rank	N	A	$(L_{act})^2$	N-1 (Number of Factors Contributing to Component Rank)
Small other safety pipe	5			X	1
Cables	5	X		X	2
Containment (BWR)	5	X			1
Connectors	5			X	1
S/G tubing	5		X		1
Turbine driven pump	4		X		1
Relays	4				0
Diesel	4		X		1
RX internals	4	X		X	2
N-2 (number of times the factor was principle contributor to rank 5 and 4 components)		3	3	4	

top-ranked components. The factors A and $(L_{act})^2$ were the principal contributors to the rank 7 out of 10 times. Had these factors not been considered, some of the surprises wouldn't have occurred. Small other safety pipe and connectors would not have had their high rank. [Cables still would have been ranked high because of N, but would not have been ranked higher than containment (BWR), which had an order of magnitude higher N.] In addition, the ranking for RPV would not have been so low (mid-rank 3).

2.3.5 Insights on Research, Regulatory and Industry Applications

The RSCAAMP methodology provides a discriminating tool, one useful beyond the ranking of S/Cs for subsequent aging research. Because it facilitates an examination of the individual factors that contribute to the risk impact of aged components, it has three other applications: 1) focusing NRC research by identifying the relevant factors for each component that need to be examined (the risk-important failure mode, aging failure rate and its mechanisms; the adequacy of surveillance/test frequencies and methods; and the adequacy of mitigation methods); 2) focusing NRC regulation toward the aging risk-significant components, and for these, the inspection/test frequencies and methods necessary to detect the risk-significant failure modes, aging mechanisms and the mitigation methods required to reduce risk; 3) providing guidance to utilities to reduce the risk contribution of their aged components [reducing N, through changes in system design, added redundancy, etc.; reducing A, through improved materials, upgraded operating conditions and environments, etc.; and reducing the actual (effective) aging interval (L_{act}) through a combination of reducing L_{ind} and increasing P_D and $P_{R/D}$].

2.4 CONCLUSIONS FROM RISK-BASED PRIORITIZATION

The prioritization of S/Cs by an expert panel using the RSCAAMP model was accomplished. In so doing, the impact on component risk contribution of aging and aging management practices, as well as the component's normal risk importance, were evaluated. For the top-ranked components (ranks 5 and 4), all of these factors were significant; no one factor was dominant.

The importance of including component aging and its management in assessing plant risk was dramatized by the unexpected findings: 1) some components not considered in PRAs were found in the top rank (rank of 5) for risk contribution (cables, connectors, small safety-related piping); 2) conversely, concrete structures and diesels that are considered in PRAs to be significant contributors to plant risk were found in the lower ranks (ranks of 2 and 4, respectively); and 3) perhaps as significant, is the number of "abundant small components" (e.g., small other safety-related piping, cables, connectors, and S/G tubes) found in the top ranking (four of the five groups ranked 5) when one might intuitively have thought of the major non-redundant, defense-in-depth structures (RPV and containment) to be top-ranked [RPV was ranked 3, containment (other) ranked 2, with only the special case of the BWR-Mk-1 containment ranked 5 (due to aging)].

Table 2.11 presents the current status of component research in the NRC Plant Aging Research Program for the ranked comonents (from Table 2.8). The table provides several conclusions:

- most components have work on-going; a few additional components have been identified as being of interest but not in the current scope (e.g., chillers, etc.)

- approximately one-half of the components ranked 1 and 2 are under study and several are considered to be of interest; a re-evaluation of their continued study/interest appears warranted

- of the components in the top two ranks, two [containment (BWR-Mk-1) and RX internals] are not being actively studied in the aging research program, and a third (S/G) is considered to be completed; consideration of their inclusion in the program appears warranted.

The panel's deliberations and findings highlight important areas for further consideration:

TABLE 2.11. Status of Aging Research on Ranked Components

Component	ΔR Rank	Research on-going	Components of Interest But Not In Scope
5.b. Small other safety pipe[a]	5	x	
10.a. Cables	5	x	
2.a. Containment (BWR)	5	0[b]	
10.b. Connectors	5	x	
6.a. S/G tube	5	--[c]	
19.b. Turbine pump	4	x	
17.a. Relay	4	x	
11. Diesel	4	x	
12. RX internals	4	0	
17.b. Breaker	3	x	
18.e. Motor operated valve	3	x	
4.c. BWR pipe (small LOCA)	3	x	
19.a. Motor pump	3	x	
5.a. Large other safety pipe[d]	3	x	
16.a. Thermostat	3	0	
24.a. Chillers	3		x
1. RPV	3	x	
25. Battery	3	x	
23. Compressor (instr. air)	3	x	
18.a. Air operated valve	2		x
30.b. DC bus	2	x	
9.a. CRDM (BWR)	2	x	
18.b. Check valve	2	x	
24.b. Fan	2		x
22. Heat exchanger	2	x	
31. Bolts	2	0	
30.a. AC bus	2	x	
18.f. Safety/relief valve	2		x
2.b. Containment (other)	2	0	
3. Other concrete structures	2		x
27. Transformer	1	x	
26.b. Inverter	1	x	
16.b. Transfer switch	1	0	
15. Snubbers	1	x	
18.c. Hydraulic valve	1	0	
21. Turbine	1	0	
16.c. Bistable	1	x	
18.d. Manual valve	1	0	
26.a. Battery charger	1	x	
29.b. Tank (atmos. pres.)	1	0	
26.c. Rectifier	1	x	
29.a. Tank (medium pres.)	1	0	
9.b. CRDM (PWR)	1	x	
[e]			
8. Pressurizer	1	0	
6.b. S/G shell	1	0	
29.c. Tank (high pres.)	1		x
4.a. RCP & SE large (LOCA)	1	x	
7. RCP casing	1	0	
13. RPV support	1	x	
28. Fuel rack	1	0	
4.b. PWR pipe (small LOCA)	1	x	

(a) 6-10 in. pipe represented by letdown and RWCV systems (see Table 2.2).
(b) 0: no activity or plan.
(c) Completed FY87.
(d) 10-24 in. pipe represented by service water piping (see Table 2.2).
(e) Components below this line were eliminated from full evaluation by the screening criterion.

- Improved data bases are required, specifically:
  --improved PRAs with up-to-date methodologies to provide component risk importances, including those of passive and accident mitigation components, and risk-significant failure modes;

  --improved aging failure data bases, to provide better aging failure data for the risk-significant failure mode.

- Improved surveillance/test methods are required to detect aging for the risk-significant failure modes in components and structures; however, once aging failures are detected, operational performance is effectively restored in many cases.

- Improved focus of research activities and regulatory actions and improved guidance to utilities regarding reduction of risk contributions of aging components can be useful outcomes of methodologies similar to the one used here.

The expert opinion process used in this study is the only way that this activity could have been accomplished; adequate data are not currently available. This study represents a starting point, not the definitive answer. As an appropriately focused research program is performed, the data generated will be useful for in-depth aging risk assessments and a refined level of prioritization of structures and components.

Finally, multiple failures of similar or dissimilar components in redundant trains was not evaluated in this prioritization effort. Performing such an evaluation and comparing the results to the current prioritization results would be a useful extension for determining the relative importance of common-cause aging. With minor development, the RSCAAMP model would be amenable to such an evaluation.

3.0 PRIORITIZATION OF STRUCTURES AND COMPONENTS USING "OTHER TECHNICAL" CRITERIA

The purpose of this section is to categorize TIRGALEX S/Cs using the other technical criteria, e.g., the importance of aging research on S/Cs to the resolution of GSIs and/or to an identified NRC/NRR user needs. S/C categorization for each of these two criteria are discussed separately; the categorization process described in Section 1.2.2.2 is followed; the results of the categorizations are tabulated and the comments of the expert panel are documented.

3.1 IMPORTANCE OF AGING RESEARCH ON S/Cs TO THE RESOLUTION OF A GSI AND ITS APPLICATION TO S/C CATEGORIZATION

This technical criterion identifies whether there would be a direct, positive impact of aging research on the resolution of NRC's GSIs. Current NRC programs are in the process of, or have completed the necessary technical steps toward solving these issues. While many issues are not affected by known aging-related phenomena, some issues may be so affected.

The list of potentially important safety issues comes from two sources: 1) a list of GSIs with elements of aging that may benefit from NPAR results (Table 3.1); and 2) additional issues discovered as a result of a pre-workshop review of all GSIs (Table 3.2).

The TIRGALEX S/Cs were initially screened to identify S/Cs for which the GSIs could directly benefit from aging research. This prescreening accounted for the technical content and available schedules associated with the resolution of the GSIs[a] and with the NRC Plant Aging Research Programs (Vora 1987). Where there was a direct technical connection (i.e., aging was involved in the issue) and the time scales appeared to be compatible, aging research on the S/Cs was identified as potentially having a direct benefit. On the other hand, some S/Cs were clearly not associated with the list of GSIs, aging was not directly related to the issue, or time scales were incompatible with their resolution; aging research on these S/Cs was classified as not beneficial. All GSI evaluations were presented to the panel for discussion and final categorization regarding whether aging research would benefit the resolution of the GSIs. A few GSIs considered relevant, but which were not identified in the current GSI resolution schedule, were also presented to the panel for consideration.

(a) Received November 1987 from R. C. Emrit, NRC/RES/DRA; found in Appendix E.

TABLE 3.1. NRC Identified Generic Safety Issues, with Elements of Aging, Potentially Benefitting from Aging Research (NUREG-1144, Rev. 1)

Number	Title
23	Reactor coolant pump seal failures
29	Bolting degradation or failures in nuclear power plant
51	Proposed requirements for improving the reliability of open cycle service water systems
55	Failures of Class 1E safety-related switchgear circuit breakers to close on demand
70	PORV and block valve reliability
84	CE PORVs
93	Steam binding of auxiliary feedwater pumps
107	Generic implications of main transformer failures
113	Dynamic qualification testing of large bore hydraulic snubbers
115	Enhancement of the reliability of Westinghouse solid state protection system
118	Tendon Anchorage failure
120	On-line testability of protection systems
124	Auxiliary feedwater system reliability
125.I.6	Valve torque limit and bypass switch settings
125.II.2	Adequacy of existing maintenance manual valves in safety-related systems
128	Electrical power reliability
130	Essential service water pump failures
132	RHR pumps inside containment
A.10	BWR feedwater nozzle cracking
A.11	Reactor vessel material toughness
A.17	Systems interaction
A.49	Pressurized thermal shock

TABLE 3.2. Additional Generic Safety Issues

Number	Title
A.3, 4, 5	Steam Generator Tube Integrity
A.40	Seismic Design Criteria
A.44	Station Blackout
B.56	Diesel Reliability
II.C.4	Reliability Engineering
II.E.4.3	Containment Integrity Check
I.F.1	Expanded QA List
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown
86	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping

3.2 IMPORTANCE OF AGING RESEARCH ON S/CS TO AN IDENTIFIED NRC USER NEED AND ITS APPLICATION TO S/C CATEGORIZATION

This technical criterion identifies components for which aging research will respond to the NRR needs expressed in the "User Need Letter".[a] In the letter, NRR expressed the need to "... know not only the effects of aging on structures, systems and components, but also the risk significance..." of the process.

3.3 RESULTS

The results and discussion for S/C categorization against the Other Technical Criteria are presented in the following sections.

3.3.1 Importance of Aging Research on S/Cs to the Resolution of a GSI

The panel was provided with screened information in the format shown in Appendix E and information on the GSI resolution schedules also shown in Appendix E. After the panelists were generally made familiar with each issue, they were asked if the technical resolution of the GSIs would benefit from S/C aging research. The panel was not asked to establish the degree of importance or the

(a) Appendix F.

expected effect of the aging research. For the GSIs that could benefit from aging research, the S/Cs that may affect the GSIs were listed. Technical comments and insights were recorded.

Tables 3.3 and 3.4 identify the GSIs discussed by the panel, and contain an indication as to whether aging research may benefit resolution of the GSIs. The specific components (from the panel's list of components in Table 2.2) for which aging research would benefit the resolution of GSIs and the panelists technical comments are also identified.

3.3.2 Importance of Aging Research on S/Cs to Resolution of User Needs

The "User Need Letter" was reviewed for components deemed to be important by NRR. Aging research on the following components would be of direct benefit to NRR.

- MOV
- Power operated relief valve (PORV)
- Block valve
- RPV
- Pumps
- Valves
- Diesels
- DC power sources

MOVs and PORVs were emphasized in several sections of the letter. In the expert panel component list, PORVs are in the category safety/relief valves; block valves are a specific application of MOVs; pumps were subdivided by prime mover; valves were subdivided into 5 categories; and DC power sources (battery, rectifier, inverter, battery charger) are all considered.

3.3.3 Conclusions

In the panel's judgement, many of the generic issues that deal with equipment performance were expected to directly benefit from aging research (schedule permitting), and that further aging research on the components identified in Section 3.3.2 would be of benefit to NRR. Table 3.5 shows the components' ranking based on risk (from Table 2.8), S/Cs for which aging research would benefit the resolution of a GSI and/or a NRC/NRR user need (from Tables 3.3 and 3.4), and those S/Cs for which current aging research is not already on-going (from Table 2.11). Only for reactor internals (meeting GSI and user needs criteria), pressurizer (GSIs), and bolts (GSI) and hydraulic and manual valves (GSI and user needs) is aging research not already on-going.

TABLE 3.3. NRC-Identified GSIs with Elements of Aging Potentially Benefitting from Aging Research

Number	Title	Will the GSI Benefit from Aging Research	Component Research That Will Benefit from the GSI	Comments
23	Reactor coolant pump seal failures	No	--	Seals were not discussed within PRA or TIRGALEX
29	Bolting degradation or failures in nuclear plant	Yes	31. Bolts	
51	Proposed requirements for improving the reliability of open cycle service water systems	Yes	22. Heat exchanger	Biofouling is an aging issue
55	Failure of class 1E safety-related switchgear circuit breakers to close on demand	Yes	17.a. Relay b. Circuit breakers	
70	PORV and block valve reliability	No		Operator error and control room design issue
84	CE PORVs	No		Design issue
93	Steam binding of auxiliary feedwater pumps	Yes	18.b. Check valves	Check valve back leakage may be affected by aging
107				
113	Dynamic qualification testing of large bore hydraulic snubbers	Yes	Snubbers	Snubbers have a high "A"
115	Enhancement of the reliability of Westinghouse solid state protection system	No		Electric design issue
118	Tendon anchorage failure	Yes	3. Category 1 structures	
120	On-line testability of protection systems	No		RPS doesn't have testing capability
124	Auxiliary feedwater system reliability	Yes	19.b. Turbine driver pump 18. All valve types	Aging is not considered in system reliability studies
125.1.6	Valve torque limit and bypass switch settings	Yes	18.e. MOV	Important because the probability of finding a problem is low
125.2.2	Adequacy of existing maintenance of manual valves in safety-related systems	Yes	18.d. Manual valve	Maintenance can mitigate aging

TABLE 3.3. (contd)

Number	Title	Will the GSI Benefit from Aging Research	Component Research That Will Benefit from the GSI	Comments
128	Electrical power reliability	Yes	17.a.b; 10.a.b; 30.a.b.	Aging is very important to this GSI
130	Essential service water pump failures	Yes	19.a. Motor driver pumps	Service water pumps exhibit a very high fraction of aging failures (f)
132	RHR pumps inside containment	?		No comment, no information on issue
A.10	BWR feedwater nozzle cracking	Yes	4. Reactor coolant piping and safe ends	
A.11	Reactor vessel material toughness	Yes	1. RPV	Vessel can be annealed
A.17	Systems interaction	No		Mathematical analysis is currently performed to model this
A.49	Pressurized thermal shock	Yes	1. RPV; 8. Pressurizer	

TABLE 3.4. Additional GSIs with Elements of Aging Potentially Benefitting from Aging Research

Number	Title	Will the GSI Benefit from Aging Research	Component Research That Will Benefit from the GSI	Comments
A.3, 4, 5	Steam generator tube integrity	Yes	G.a. Steam generator tube	
A.40	Seismic design criteria	No		Studies criteria, not change in fragility
2.C.4	Reliability engineering	No		
2.E.4.3	Containment integrity check	No		Checks for gross openings only
1.F.1	Expanded QA list	No		QA is probably not aging-related
79	Unanalyzed reactor vessel thermal stress during natural convection cooldown	Yes	1. RPV	
86	Long range plan for dealing with stress corrosion cracking in BWR piping	Yes	4. Reactor coolant piping	
A.44	Station blackout	Yes	10, 11, 17, 30	
B.56	Diesel generator reliability	Yes	11	

TABLE 3.5. Status of Aging Research on S/Cs Ranked by Risk Importance and Other Technical Criteria

Component	ΔR Rank	Research on-going	Components of Interest But Not in Scope
5.b. Small other safety pipe[a]	5	x	
10.a. Cables	5	x	
2.a. Containment (BWR)	5	0[b]	
10.b. Connectors	5	x	
6.a. S/G tube	5	--[c]	
19.b. Turbine pump	4	x	
17.a. Relay	4	x	
11. Diesel	4	x	
12. RX internals	4	0	
17.b. Breaker	3	x	
18.e. Motor operated valve	3	x	
4.c. BWR pipe (small LOCA)	3	x	
19.a. Motor pump	3	x	
5.a. Large other safety pipe[d]	3	x	
16.a. Thermostat	3	0	
24.a. Chillers	3		x
1. RPV	3	x	
25. Battery	3	x	
23. Compressor (instr. air)	3	x	
18.a. Air operated valve	2		x
30.b. DC bus	2	x	
9.a. CRDM (BWR)	2	x	
18.b. Check valve	2	x	
24.b. Fan	2		x
22. Heat exchanger	2	x	
31. Bolts	2	0	
30.a. AC bus	2	x	
18.f. Safety/relief valve	2		x
2.b. Containment (other)	2	0	
3. Other concrete structures	2		x
27. Transformer	1	x	
26.b. Inverter	1	x	
16.b. Transfer switch	1	0	
15. Snubbers	1	x	
18.c. Hydraulic valve	1	0	
21. Turbine	1	0	
16.c. Bistable	1	x	
18.d. Manual valve	1	0	
26.a. Battery charger	1	x	
29.b. Tank (atmos. pres.)	1	0	
26.c. Rectifier	1	x	
29.a. Tank (medium pres.)	1	0	
9.b. CRDM (PWR)	1	x	

[e]

Component	ΔR Rank	Research on-going	Components of Interest But Not in Scope
8. Pressurizer	1	0	
6.b. S/G shell	1	0	
29.c. Tank (high pres.)	1		x
4.a. RCP & SE large (LOCA)	1	x	
7. RCP casing	1	0	
13. RPV support	1	x	
28. Fuel rack	1	0	
4.b. PWR pipe (small LOCA)	1	x	

(a) 6-10 in. pipe represented by letdown and RWCV systems (see Table 2.2).
(b) 0: no activity or plan.
(c) Completed FY87.
(d) 10-24 in. pipe represented by service water piping (see Table 2.2).
(e) Components below this line were eliminated from full evaluation by the screening criterion.

4.0 CONCLUSIONS AND RECOMMENDATIONS

The expert panel workshop was conducted to prioritize the TIRGALEX set of nuclear power plant S/Cs for further evaluation within the NRC Plant Aging Research Program. The prioritization was primarily based upon risk-based criteria; other technical criteria were used to categorize those S/Cs for which aging research would benefit the resolution of GSIs and/or identified NRC/NRR user needs but were not used to rank the S/Cs.

From the S/C prioritization using risk-based criteria, the major conclusions from the workshop are the following: 1) the prioritization of S/Cs was accomplished by an expert panel using the multi-factor RSCAAMP methodology. Analysis of the results showed that all factors of this methodology are equally significant to an assessment of the relative risk importances of aged components. Further, the importance of including component aging and its management in plant risk assessments was dramatized by the unexpected findings in the top-ranked components. 2) Current aging research warrants reevaluation: many low-ranked S/Cs (ranks 1 and 2) are under study and others are considered to be of interest; two of the components in the top two ranks (ranks 5 and 4) are not currently being studied in the Plant Aging Research Program [containment (BWR-Mk-1) and reactor internals]. 3) the panel's deliberations and findings highlighted the need for improved PRA and aging-failure data bases, the shortfalls in current industry practices for detecting aging for risk-significant failure modes in components and structures, and the usefulness of the methodology for focusing research and regulatory actions and for providing utilities with areas to address in order to reduce the risk-contribution of their aged components. 4) This study should be viewed as a starting point for which expert opinion was required; from this study an appropriately-focused research program can generate the data needed for a more definitive answer on the relative risk-importance of aged components. 5) The explicit evaluation of the importance of aging-induced common-cause failures would be a useful extension of the methodology, with minor development, the RSCAAMP model would be amenable to such an evaluation.

From the S/C categorization using other technical criteria, the major conclusions from the workshop are as follows: 1) the S/Cs were categorized using the other technical criteria. 2) There are many S/Cs for which an aging research program could have direct benefit to the resolution of GSIs. 3) There are fewer, but still substantial numbers of S/Cs for which an aging research program could respond directly to identified user needs. 4) For all but a few of these components aging research is already on-going.

Table 3.5 identified the S/Cs prioritized by the risk-based criteria, the S/Cs upon which aging research could benefit the resolution of GSIs and User Needs, and the S/Cs for which aging research is not already on-going or of current interest.

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APPENDIX A

DERIVATION OF BASIC RSCA MODEL AND THE RANKING METHODOLOGY

APPENDIX A

DERIVATION OF BASIC RSCA MODEL AND THE RANKING METHODOLOGY

The enclosed material was taken from an introductory workshop presentation given by A. Wolford.

RSCA Methodology

- Unavailability, hence risk, is time dependent due to:
  - Normal variations arising from constant failure rate with repair (standard sawtooth behavior)
  - Increasing failure rate due to aging degradation
- We are interested in quantifying the increase in risk due to aging as separated from effects without aging
- Define:

  R_o = Total Risk including aging effects
  R = Risk without aging contribution
  q_o = Total instantaneous unavailability including aging
  q = Instantaneous unavailability without aging contribution

RSCA Methodology
Highlights of Derivation

- The risk change due to aging may be expressed as

$$\Delta R = (R - R_o) = \int \frac{\partial (R - R_o)}{\partial t} \, dt^{*}$$

- The derivatives on the right hand side, with use of the chain rule, may be expressed as

$$\frac{\partial R}{\partial t} = \frac{\partial R}{\partial q} \frac{\partial q}{\partial t} \qquad (1)$$

and

$$\frac{\partial R_o}{\partial t} = \frac{\partial R_o}{\partial q_o} \frac{\partial q_o}{\partial t} \qquad (2)$$

* R, q both time dep., notation will be relaxed

RSCA Methodology

Highlights of Derivation

- We may write

$$\frac{\partial (R - R)}{\partial t} = \frac{\partial R}{\partial q}\left(\frac{\partial q}{\partial t} - \frac{\partial q_o}{\partial t}\right) + \left(\frac{\partial R}{\partial q} - \frac{\partial R_o}{\partial q_o}\right)\frac{\partial q_o}{\partial t} \qquad (3)$$

- The first term on the right hand side is called the Risk Sensitivity due to a Component's Aging. It is comprised of the normal risk importance of the component and the additional rate of change in component unavailability due to aging-driven degradation

- The second term is the contribution from the nonaging (constant failure rate) change in component unavailabilities

- Generally, the second term will be small compared to the first

RSCA Methodology
Highlights of Derivation

- A specific model of unavailability is required for explicit evaluation of Equation (3)

- Employing the Linear Aging Model to describe increasing failure rate,

$$\lambda(t) = at + \lambda_o \tag{4}$$

- The small availability approximation allows

$$\frac{\partial q}{\partial t} - \frac{\partial q_o}{\partial t} \approx \lambda(t) - \lambda_o = at \tag{5}$$

Where an estimate of "a" is given by Equation (12)

- The final form for the basic RSCA equation becomes

$$\frac{\partial (R - R_o)}{\partial t} = \frac{\partial R}{\partial q} \cdot at \tag{6}$$

RSCA Methodology
Working Definitions of Terms

- Risk
  Generally taken to be core melt frequency, in PRA sense

- Normal Risk Importance..*is not Risk Contribution*
  Also called Birnbaum Measure or Reliability Importance. Is a measure of the critical relationship to the system in which it resides. It considers both system design (eg. redundancy) and the components reliability.

- Instantaneous Unavailability
  The probability that the component is not operating at the instant of time, t

- Linear Aging Model
  Time dependent failure rate modeled as linearly dependent on time

RSCA Methodology
Working Definitions of Terms

- Aging Contribution
  In the development of RSCA we have made the working hypothesis that figures of merit for plants, systems and components (eg. Risk) may be separated into fractions which are due to (systematic) aging and to those which are random in nature. The aging contribution is the fraction of some system variable which is modeled as aging-caused

- Aging Rate
  Slope of the failure rate-time function (LAM) above

- Sensitivity
  A derivative

Enhancements to the RSCA Methodology

- Agregation of component importances
  How component's RSCA's may be combined within and across plants

- Separation into Factors
  How the RSCA equation is separated into physically-meaningful factors

- Incorporation of Current Aging Management Adequacy Criteria
  Method for inclusion of quantitative criteria addressing the adequacy of current aging management practices for maintaining aging increased risk at acceptable levels

Enhancements to the RSCA Methodology

Agregation of Importances

- Generalization of Equation (6) from a single component to groups of components involves only the following trivial summation

$$\frac{\partial \Delta R^G}{\partial t} = \sum_{i \in G} N_i a_i t \qquad (7)$$

- Where we have made the following notation changes

$$\Delta R^G = \sum_{i \in G} (R_i - R_{0i})$$

$$N_i = \frac{\partial R_i}{\partial q_i}$$

Enhancements to the RSCA Methodology

Separation of Factors

- We wish to prioritize component groups, specifically the TIRGALEX groups, by Aging Risk Increase, hence Equation (7) should be integrated, thus:

$$\Delta R^{G} = \int \sum_{i \in G} N_i a_i t \, dt \qquad (8)$$

which results in

$$\Delta R^{G} = \sum_{i \in G} N_i a_i \frac{L^2}{2} \qquad (9)$$

Enhancements to the RSCA Methodology
Separation into Factors

- We subdivide the right hand side of Equation (9) into three (3) factors, which are physically interpretable

$$\Delta R^G = N^G \langle a\rangle^G \Lambda^G \qquad (10)$$

- Meanings

  - $N^G = \sum_{i \in G} N_i$ = Normal Risk Importance for the Group
  - $\langle a\rangle^G = \dfrac{\sum_{i \in G} N_i a_i}{\sum_{i \in G} N_i}$ = Average Effective Aging Rate for the group
  - $\Lambda^G = \dfrac{L^2}{2}$ = Aging Exposure Period Factor
    Integration causes power of two

Implementation of the RSCA Methodology in an Expert Panel framework

- Generic estimates were taken from the four plant study by a modified median
- Standard median was taken for odd samples, geometric mean central values for even samples
- This modified median was thought to provide the least biased generic estimates from the limited four plant survey

Implementation of the RSCA Methodology in an Expert Panel Framework

- A ranking process was developed to facilitate data handling and presentation which also allows for less cumbersome manipulation and interpolation

- The ranking scheme, however, is based on a logrithmic scale, and the *subtleties* should be well understood.

- The ranking process ranks each of the factors to the nearest integral base 10 log.

  The Group with the maximum rank is scaled, by adding a constant, so that the rank is equal to 5, the maximum rank for each factor.

  Each group is also scaled by the same constant

- In Equation form: $RANK(zi) = \log(zi) + Cz$ (11)

  where $Cz = 5 - \log\{Max(zi)\}$

ESTIMATES OF AGING FAILURE RATE (A)

An estimate of the aging failure rate, A. may be obtained by the method of moments which is:

$$A = \frac{4}{3}\frac{\lambda f}{T_A} \tag{12}$$

where f = The number fraction of failures determined to be caused by aging related stressors

λ = The constant average failure rate. Generic values are used in this study

T_A = The mean time to aging failure

APPENDIX B

DATA TABLES SUPPORTING THE PRECALCULATED RSCA FACTORS

APPENDIX B

DATA TABLES SUPPORTING THE PRECALCULATED RSCA FACTORS

TABLE B.1. Four-Plant Component Data Summary: Number of Components by Component Type

Component Group	Component Type	Number of Components: ANO1	Grand Gulf	Calvert Cliffs	Oconee	Total
Pumps	Motor drive pump	9	9	10	11	39
	Turbine driven pump	2		2	1	5
Valves	Air operated valve			2	4	6
	Check valve	37	21	21	21	100
	Hydraulic valve	4				4
	Manual valve	85	31	31	15	162
	Motor operated valve	55	47	22	20	145
	Safety/relief valve	5	1			7
Diesels	Diesel generator	2	3	2		7
Electrical	Thermostat	1				2
	Circuit breaker	65		8	4	77
	Load/relay switch			4		4
	Transfer switch	1				1
	Bistable trip unit	3				3
Power	Battery	2	3	3	2	10
	Battery charger	2				2
	Inverter	4				4
	Transformer	14				14
	DC bus	7				7
	AC bus	11				1
	Rectifier	12				12
Heat exchanger	Turbine		1			1
	Air conditioner			2		2
	Heat exchanger	6				6
	Motor driven fan	7				7
Structural	Tank	3				3
Totals		336	115	107	78	638

TABLE B.2. Four-Plant Component of Data Summary: Number of Components by System

Component Group	System	ANO1	Grand Gulf	Calvert Cliffs	Oconee	Total
Pumps	AC emergency power	1			1	2
	Auxiliary feedwater system	3		2	2	7
	High pressure core spray		1			1
	High pressure injection	3		2	2	7
	Low pressure coolant injection		3			3
	Low pressure core spray		1			1
	Low pressure injection	2		2	3	7
	Reactor core isolation cooling		1			1
	Service water	2	3	6	4	15
Valves	Auxiliary feedwater system	39		22	24	85
	High pressure core spray		6			6
	High pressure recirculation	2				2
	High pressure injection	42		16	12	70
	Low pressure coolant injection		20			20
	Low pressure core spray		5			5
	Low pressure recirculation	4				4
	Low pressure injection	25		13	22	60
	Primary pressure relief	2	1			3
	Reactor core isolation cooling		15			15
	Residual heat removal		32			32
	Service water	52	19	22		93
	Suppression pool monitoring		2			2
	Emergency cooking	14				14
	Core flood	6				6
Diesel	AC emergency power	1	3	2		6
	Auxiliary feedwater system	1				1
Electrical	AC emergency power	13				13
	Auxiliary feedwater system	2				2
	DC emergency power	39				39
	Engineered safety feature act.	3				3
	Reactor protection	12		12	4	28
	Service water					1
Power	AC emergency power	12				12
	DC emergency power	35	3	3	2	43
	Reactor protection	4				4
Heat Exchanger	Auxiliary feedwater system	1				1
	Low pressure injection	2				2
	Service water	7		2		9
	Emergency cooling	4				4
	Core flood	2				2
Totals		335	117	104	76	631

TABLE B.3. Four-Plant Component Risk Importance Data by Component Type

Component: TIRGALEX Group	Component: PRA	Component of Importances: ANO1	Grand Gulf	Calvert Cliffs	Oconee	Generic Importances
11. Diesels	Diesel generator	9.2E-4	2.2E-4	3.7E-3		9.2E-4
16. I&C	a. Thermostat	6.0E-3				6.0E-3
	b. Transfer switch	4.7E-6				4.7E-6
	c. Bistable trip unit	1.2E-5				1.2E-5
17. Switchgear/ Relays	a. Relay (Load)			4.8E-2		4.8E-2
	b. Circuit breaker	7.8E-2		7.2E-2	1.8E-3	7.2E-2
18. Valves	a. Air operate valve			3.2E-2	2.6E-4	3.2E-2
	b. Check valve	9.3E-2	2.7E-3	1.3E-1	3.4E-3	1.8E-2
	c. Hydraulic valve	3.8E-4				3.8E-4
	d. Manual valve	1.6E-1	6.9E-3	1.2E-0	5.1E-3	3.3E-2
	e. Motor operated valve	7.1E-2	9.3E-3	5.0E-2	2.7E-3	2.2E-2
	f. Safety/relief valve	6.9E-3	6.1E-5		3.5E-4	3.5E-4
19. Pumps	a. Motor driven pump	8.9E-3	2.1E-3	5.1E-3	2.3E-2	6.7E-3
	b. Turbine drive pump	9.3E-3		3.2E-2	2.0E-6	9.3E-3
21. Turbine			1.0E-4			1.0E-4
22. Heat exchanger	Heat exchanger	6.4E-3				6.4E-3
23. Compressor	Air conditioner			7.7E-4		7.7E-4
24. Fans/ chiller	Fan	7.6E-3				7.6E-3
25. Battery	Battery	3.9E-3	5.7E-4	1.5E-3	1.7E-4	9.2E-4
26. Battery charger inverter	a. Battery charger	1.1E-4				1.1E-4
	b. Inverter	4.2E-4				4.2E-4
	c. Rectifier	4.7E-4				4.7E-4

TABLE B.3. (contd)

Component		Component of Importances				
TIRGALEX Group	PRA	ANO1	Grand Gulf	Calvert Cliffs	Oconee	Generic Importances
27. Transformer	Transformer	1.2E-2				1.2E-2
29. Accumulator/ Tanks	Tank	2.7E-2				2.7E-2
30. AC/DC bus	a. AC bus	4.3E-2				4.3E-2
	b. DC bus	1.1E-1				1.1E-1

NOTE: Median importances are used when an odd number importance values were available. The geometric center (mean) of the middle two importances was used when an even number of importance values was available.

TABLE B.4a. System Risk Importances and the Importances of Components Within the System (ANO1)

	Normal Risk Importances (N) ANO1																		
Component/System	EPS AC	EPS-DC	AFNS	ESFAS	HPCS	HPR	HPIS	LPCIS	LPCS	LPR	LPIS	PPR	RCICS	RPS	RHR	SWS	SPMS	ECS	CFS
Air conditioner																			
Thermostat																6E-3			
Circuit breaker	5E-2	9E-3	1E-3											1E-2					
Motor - driven fan																1E-3		6E-3	
Diesel generator	3E-4		7E-4																
Heat exchanger											2E-4					6E-3			
Invertor		4E-4																	
Relay																			
Motor - driven pump	3E-4		9E-5				3E-5				2E-4					8E-3			
Turbine - driven pump			6E-3				3E-3												
Rectifier		4E-4												3E-5					
Transfer switch		5E-6																	
Transformer	1E-2	5E-4																	
Tank			3E-2																2E-4
Bistable trip unit				1E-5															
DC - bus		1E-1																	
AC - bus	4E-2	4E-5																	
Battery		4E-3																	
Air - operated valve																			
Check valve			8E-2				3E-3				4E-4					1E-2			2E-4
Hydraulic valve											4E-4								
Manual (hand) operated valve			4E-2			1E-4	1E-2				2E-2					6E-2		3E-2	
Motor operated valve		2E-2					3E-3			4E-4	4E-3					5E-2			2E-4
Safety/relief valve			6E-3									3E-4							2E-4
Battery charger		8E-5																	
Turbine																			
Sum of System 'N'	1E-1	1E-1	2E-1	1E-5		1E-4	2E-2			4E-4	3E-2	3E-4		1E-2		1E-1		4E-2	8E-4

TABLE B.4b. [contd (Grand Gulf)]

Component/System	Normal Risk Importances (N) Grand Gulf																		
	EPS AC	EPS-DC	AFNS	ESFAS	HPCS	HPR	HPIS	LPCIS	LPCS	LPR	LPIS	PPR	RCICS	RPS	RHR	SWS	SPMS	ECS	CFS
Air conditioner																			
Thermostat																			
Circuit breaker																			
Motor - driven fan																			
Diesel generator	2E-4																		
Heat exchanger																			
Invertor																			
Relay																			
Motor - driven pump					7E-5			6E-4	1E-5				1E-4			1E-3			
Turbine - driven pump																			
Rectifier																			
Transfer switch																			
Transformer																			
Tank																			
Bistable trip unit																			
DC - bus																			
AC - bus																			
Battery		6E-4																	
Air - operated valve																			
Check valve					2E-4			6E-4	3E-5				5E-4		4E-5	2E-3			
Hydraulic valve																			
Manual (hand) operated valve					7E-5			6E-4	1E-5				2E-4		4E-3	2E-3			
Motor operated valve					1E-4			6E-4	3E-5				8E-4		5E-3	3E-3	9E-5		
Safety/relief valve												6E-5							
Battery charger																			
Turbine																			
Sum of System	2E-4	6E-4			4E-4			2E-3	8E-5			6E-5	2E-3		9E-3	8E-3	9E-5		

TABLE B.4c. [contd (Calvert Cliffs)]

Component/System	Normal Risk Importances (N) Calvert Cliffs																		
	EPS AC	EPS-DC	AFNS	ESFAS	HPCS	HPR	HPIS	LPCIS	LPCS	LPR	LPIS	PPR	RCICS	RPS	RHR	SWS	SPMS	ECS	CFS
Air conditioner																8E-4			
Thermostat																			
Circuit breaker														7E-2					
Motor - driven fan																			
Diesel generator																			
Heat exchanger																			
Invertor																			
Relay														5E-2					
Motor - driven pump							1E-3				4E-6					4E-3			
Turbine - driven pump			3E-2																
Rectifier																			
Transfer switch																			
Transformer																			
Tank																			
Bistable trip unit																			
DC - bus																			
AC - bus																			
Battery		2E-3																	
Air - operated valve			3E-2																
Check valve			1E-1				3E-3				8E-6					3E-4			
Hydraulic valve																			
Manual (hand) operated valve			1E+0				2E-3				1E-5					1E-2			
Motor operated valve			3E-2				1E-2				3E-4					7E-3			
Safety/relief valve																			
Battery charger																			
Turbine																			
Sum system	4E-3	2E-3	1E+0				2E-2				3E-4			1E-1		2E-2			

TABLE B.5. Average Age (T_A) Matrix by Component and System

	EPS-AC	AFWS	EPS-DC	ESFAS	HPCS	HPR	HPIS	LPCIS	LPCS	LPR	LPIS	PPR	RCICS	RPS	RHR	SWS	SPMS	ECS	CFS
Air conditioner																10.57			
Thermostat																4.9			
Circuit breaker	6.4	6.4	6.4											6.4					
Motor - driven fan																8.51		8.04	
Diesel generator	6.3	6.3																	
Heat exchanger											11.53					11.53			
Invertor			4.86																
Relay														8.2					
Motor - driven pump	6.5	6.5			4.2		4.2	4.2	4.2		4.2					8.9			
Turbine - driven pump		9					4.2												
Rectifier			9.34											9.34					
Transfer switch			1.33																
Transformer	9.34		9.34																
Tank		15.73																	
Bistable trip unit				4.9															
DC - bus			8.02																
AC - bus	8.02		8.02																
Battery			9.4																
Air - operated valve		4.9																	
Check valve		6.5			7.4		6.5	6.5	6.5		6.5		6.5		6.5	8.9	6.5		
Hydraulic valve								5.9			0.3								
Manual (hand) operated valve		6.5			6.5	6.5	6.5	6.5	6.5		6.5		6.5		6.5	7.6			6.5
Motor operated valve		5.9			9.1		5.9	5.9	5.9	5.9	5.9		5.9		5.9	8.5	5.9		
Safety/relief valve		7.6										4.3							
Battery charger			9.01										8.9						
Turbine																			

TABLE B.6. Aging Fraction (f) Matrix by Components and System

	Aging Survey Analysis Data: Aging Fraction (f) (No System Dependencies)	Aging Survey Analysis Data: Aging Fractions (f) (with System Dependencies)			Repeated Failure Cause Analysis Fractions (f)[f,h,j]			
		CCW	IE	RPS	AFW	IE	HPIS	SWS
Air conditioner	0.6[a]							
Thermostat	0.25[b]							
Circuit breaker	0.26							
Motor - driven fan	0.6[a]							
Diesel generator	0.25							
Heat exchanger	0.52							
Invertor	0.24							
Relay	0.25							
Motor - driven pump[g]	0.45	0.75			0.48		0.47	0.77
Turbine - driven pump[g]	0.45				0.47			
Rectifier	0.27[c]							
Transfer switch	0.10	0.19	0.19					
Transformer	0.16							
Tank	0.42[d]							
Bistable trip unit	0.23[e]							
DC - bus	0.09							
AC - bus	0.09							
Battery	0.32							
Air - operated valve	0.49				0.63			0.77
Check valve[i]	0.85				0.87		0.83	0.87
Hydraulic valve	0.49				0.63			0.77
Manual (hand) operated valve	0.49				0.75			
Motor operated valve	0.49				0.51		0.46	0.39
Safety/relief valve	0.49							
Battery charger	0.32							
Turbine	0.16							
Motor	0.36							
Valve operator	0.25							

TABLE B.6. (contd)

(a) Used blower/compressor.
(b) Used instrumentation: indicator/recorder.
(c) Used instrumentation: electronic power supply.
(d) Used vessel.
(e) Used instrumentation: controller.
(f) Used the lower bound from the repeated failure cause because it excluded unknown failures.
(g) Few turbine and motor failures are reported relative to pump failures so the pump aging rate is appropriate for motor- and turbine-driven pumps.
(h) The CTIS system data is unique, but the system is not included in PRA, so it is not considered here.
(i) Check valves generally have very simple or no operator, so the failure cause analysis data ($f \sim 0.85$) was used.
(j) The f-values of the more detailed report failure cause analysis portion of the study are noted as being higher than the f-values of the aging survey analysis, but are only used for specific systems. The system independent values of the aging survey analysis are generally used.

TABLE B.7. Component Failure Rates and Failure Rate Doubling Times

	Failure Rate (Per Stby Hr)	Aging Rate (Per hr per year)	Doubling Time (year)
11. Diesel generator	4.0E-5	1.6E-6	25.0
16. a. Thermostat	3.0E-6	1.5E-7	20.2
b. Transfer switch	3.0E-6	2.3E-7	13.1
c. Bistable trip unit	3.0E-6	1.4E-7	21.9
17. a. Relay (load)	3.0E-6	9.1E-8	32.9
b. Circuit breaker	4.0E-6	1.6E-8	250.3
18. a. Air operate valve	4.0E-6	4.0E-7	10.0
b. Check valve	3.0E-6	3.8E-7	8.0
c. Hydraulic valve	4.0E-6	6.5E-6	0.6
d. Manual valve	3.0E-7	2.2E-8	13.8
e. Motor operated valve	4.0E-6	2.6E-7	15.2
f. Safety/relief valve	1.0E-5	6.7E-7	14.8
19. a. Motor driven pump	4.0E-6	2.2E-7	18.4
b. Turbine drive pump	4.0E-5	2.7E-6	14.6
21. Turbine	2.0E-4[a]	3.7E-6	54.8
22. Heat exchanger	3.0E-6	1.4E-7	21.9
23. Air conditioner	1.0E-5	5.6E-7	17.9
24. Fan	1.0E-6	2.1E-7	4.9
25. Battery	1.0E-6	3.4E-8	29.2
26. a. Battery charger	1.0E-6	3.5E-8	28.3
b. Invertor	1.0E-4	4.9E-6	20.4
c. Rectifier	3.0E-6	8.7E-8	34.6
27. Transformer	1.0E-6	1.7E-8	58.4
29. Tank	1.0E-9[b]	2.7E-11	37.0
30. a. AC bus	1.0E-8	1.1E-11	909.1
b. DC bus	1.0E-8	1.1E-10	89.3

NOTE: IREP Procedures Guide Generic Database was used to obtain failure rates except as noted. Monthly tests were assumed when necessary.

(a) An IEEE-500 composite of steam driven condensing turbines was used for turbine failure rate.

(b) The number used in the ANO-1 PRA. Probably an upper bound.

TABLE B.8. Component N and A Values by System

Plant: ANO1 Unit 1

Comp	System	N	a	Count
Thermostat	Service water	6.03E-03	1.34E-03	2
Circuit breaker	Emer power system - AC	5.47E-02	1.37E-04	13
Circuit breaker	Auxiliary feedwater system	1.30E-03	1.38E-04	2
Circuit breaker	Emer power system - DC	8.51E-03	1.38E-04	38
Circuit breaker	Reactor protection system	1.36E-02	1.37E-04	12
Motor-driven fan	Service water	1.14E-03	9.90E-04	3
Motor-driven fan	Emergency cooling	6.45E-03	1.90E-03	4
Diesel generator	Emer power system - AC	2.57E-04	1.39E-02	1
Diesel generator	Auxiliary feedwater system	6.67E-04	1.39E-02	1
Heat exchanger	Low press injection system	1.90E-04	1.21E-03	2
Heat exchanger	Service water	6.24E-03	1.19E-03	4
Invertor	Emer power system - DC	4.25E-04	4.33E-02	4
Motor-driven pump	Emer power system - AC	2.50E-04	2.44E-03	1
Motor-driven pump	Auxiliary feedwater system	9.32E-04	2.42E-03	2
Motor-driven pump	High press injection system	3.22E-05	3.73E-03	2
Motor-driven pump	Low press injection system	1.86E-04	3.76E-03	2
Motor-driven pump	Service water	7.54E-03	1.77E-03	2
Turbine-driven pump	Auxiliary feedwater system	6.42E-03	1.75E-02	1
Turbine-driven pump	High press injection system	2.92E-03	3.76E-02	1
Rectifier	Emer power system - DC	4.44E-04	7.66E-04	8
Rectifier	Reactor protection system	2.84E-05	7.04E-04	4
Transfer switch	Emer power system - DC	4.70E-06	2.13E-03	1
Transformer	Emer power system - AC	1.14E-02	1.50E-04	2
Transformer	Emer power system - DC	4.67E-04	1.50E-04	12
Tank	Auxiliary feedwater system	2.70E-02	3.70E-07	1
Tank	Core flood	1.50E-04	0.00E+00	2
Bistable	Emer. safety feature act system	1.15E-05	8.66E-04	3
DC bus	Emer power system - DC	1.14E-01	9.62E-07	7
AC bus	Emer power system - AC	4.29E-02	0.00E+00	10
AC bus	Emer power system - DC	3.50E-05	0.00E+00	1
Battery	Emer power system - DC	3.90E-03	3.00E-04	2
Check valve	Auxiliary feedwater system	7.82E-02	3.44E-03	12
Check valve	High press injection system	3.26E-03	3.44E-03	10
Check valve	Low press injection system	3.75E-04	3.44E-03	10
Check valve	Service water	1.13E-02	2.51E-03	3
Check valve	Core flood	1.50E-04	3.47E-03	2
Hydraulic valve	Low press injection system	3.77E-04	5.73E-02	4
Manual operated valve	Auxiliary feedwater system	4.04E-02	1.98E-04	5
Manual operated valve	High press recirc	1.35E-04	2.22E-04	2
Manual operated valve	High press injection system	1.36E-02	1.99E-04	25
Manual operated valve	Low press injection system	2.46E-02	1.98E-04	7
Manual operated valve	Service water	5.91E-02	1.69E-04	32
Manual operated valve	Emergency cooling	2.53E-02	1.98E-04	14
Motor operated valve	Auxiliary feedwater system	1.61E-02	2.91E-03	21
Motor operated valve	High press injection system	3.11E-03	2.91E-03	7
Motor operated valve	Low press recirculation	3.81E-04	2.91E-03	4

TABLE B.8. (contd)

Plant: ANO1 Unit 1 (contd)

Comp	System	N	a	Count
Motor operated valve	Low press injection system	4.39E-03	2.91E-03	4
Motor operated valve	Service water	4.73E-02	2.02E-03	17
Motor operated valve	Core flood	1.50E-04	2.93E-03	2
Safety/relief valve	Auxiliary feedwater system	6.40E-03	5.65E-03	1
Safety/relief valve	Primary pressure relief	3.00E-04	1.00E-02	2
Safety/relief valve	Core flood	1.50E-04	1.00E-02	2
Battery charger	Emer power system - DC	7.50E-05	2.67E-04	1

Plant: Grand Gulf

Comp	System	N	a	Count
Diesel generator	Emer power system - AC	2.21E-04	1.39E-02	3
Motor-driven pump	High press core spray	6.50E-05	3.69E-03	1
Motor-driven pump	Low press cool system	5.56E-04	3.76E-03	3
Motor-driven pump	Low press core spray	1.40E-05	3.57E-03	1
Motor-driven pump	Resid core isol cool system	1.00E-04	3.80E-03	1
Motor-driven pump	Service water	1.37E-03	1.77E-03	3
Battery	Emer power system - DC	5.71E-04	2.98E-04	3
Check valve	High press core spray	1.95E-04	3.03E-03	3
Check valve	Low press cool system	5.89E-04	3.44E-03	6
Check valve	Low press core spray	2.80E-05	3.57E-03	2
Check valve	Resid core isol cool system	5.00E-04	3.44E-03	5
Check valve	Residual heat removal	3.80E-05	3.42E-03	2
Check valve	Service water	1.37E-03	2.51E-03	3
Manual operated valve	High press core spray	6.50E-05	1.54E-04	1
Manual operated valve	Low press cool system	5.89E-04	2.04E-04	6
Manual operated valve	Low press core spray	1.40E-05	0.00E+00	1
Manual operated valve	Resid core isol cool system	2.00E-04	2.00E-04	2
Manual operated valve	Residual heat removal	4.40E-03	1.98E-04	12
Manual operated valve	Service water	1.67E-03	1.68E-04	9
Motor operated valve	High press core spray	1.30E-04	1.92E-03	2
MMotor operated valve	Low press cool system	6.07E-04	2.92E-03	8
Motor operated valve	Low press core spray	2.80E-05	2.86E-03	2
Motor operated valve	Resid core isol cool system	8.00E-04	2.91E-03	8
Motor operated valve	Residual heat removal	4.59E-03	2.91E-03	18
Motor operated valve	Service water	3.02E-03	2.02E-03	7
Motor operated valve	Supp pool monitoring	8.80E-05	2.95E-03	2
Safety/relief valve	Primary pressure relief	6.10E-05	1.00E-02	1

Plant: Oconee

Comp	System	N	a	Count
Circuit breaker	Reactor protection system	1.80E-03	1.39E-04	4
Motor-driven pump	Emer power system - AC	7.00E-05	3.71E-03	1
Motor-driven pump	Auxiliary feedwater system	7.00E-05	2.43E-03	1

TABLE B.8. (contd)

Plant: Oconee (contd)

Comp	System	N	a	Count
Motor-driven pump	High press injection system	2.92E-04	3.76E-03	2
Motor-driven pump	Low press injection system	1.46E-03	3.76E-03	3
Motor-driven pump	Service water	2.15E-02	1.77E-03	4
Turbine-driven pump	Auxiliary feedwater system	2.00E--6	2.00E-02	1
Battery	Emer power system - DC	1.72E-04	2.91E-04	2
Air operated valve	Auxiliary feedwater system	2.64E-04	3.52E-03	4
Check valve	Auxiliary feedwater system	8.00E-04	3.44E-03	8
Check valve	High press injection system	3.18E-04	3.43E-03	3
Check valve	Low press injection system	2.28E-03	3.44E-03	10
Manual operated valve	Auxiliary feedwater system	1.48E-04	2.03E-04	6
Manual operated valve	High press injection system	4.01E-03	1.97E-04	5
Manual operated valve	Low press injection system	9.20E-04	1.96E-04	4
Motor operated valve	Auxiliary feedwater system	1.48E-04	2.91E-03	6
Motor operated valve	High press injection system	4.58E-04	2.90E-03	4
Motor operated valve	Low press injection system	1.82E-03	2.91E-03	8
Motor operated valve	Supp pool monitoring	2.80E-04	2.93E-03	2

Plant: Calvert Cliffs

Comp	System	N	a	Count
Air conditioner	Service water	7.70E-04	4.91E-03	2
Circuit breaker	Reactor protection system	7.20E-02	1.37E-04	8
Diesel generator	Emer power system - AC	3.70E-03	1.39E-02	2
Load/relay unit	Reactor protection system	4.80E-02	8.02E-04	4
Motor-driven pump	High press injection system	1.11E-03	3.76E-03	2
Motor-driven pump	Low press injection system	4.20E-06	4.76E-03	2
Motor-driven pump	Service water	3.97E-03	1.77E-03	6
Turbine-driven pump	Auxiliary feedwater system	3.20E-02	1.75E-02	2
Battery	Emer power system - DC	1.50E-03	2.99E-04	3
Air operated valve	Auxiliary feedwater system	3.20E-02	3.51E-03	2
Check valve	Auxiliary feedwater system	1.28E-01	3.44E-03	8
Check valve	High press injection system	3.35E-03	3.44E-03	6
Check valve	Low press injection system	8.40E-06	3.57E-03	4
Check valve	Service water	3.10E-04	2.52E-03	1
Check valve	Supp pool monitoring	7.70E-04	3.44E-03	2
Manual operated valve	Auxiliary feedwater system	1.21E+00	1.98E-04	10
Manual operated valve	High press injection system	2.22E-03	1.98E-04	4
Manual operated valve	Low press injection system	1.26E-05	0.00E+00	6
Manual operated valve	Service water	9.94E-03	1.70E-04	11
Motor operated valve	Auxiliary feedwater system	3.20E-02	2.91E-03	2
Motor operated valve	High press injection system	1.04E-02	2.91E-03	6
Motor operated valve	Low press injection system	3.30E-04	2.91E-03	3
Motor operated valve	Service water	6.71E-03	2.02E-03	10
Motor operated valve	Supp pool monitoring	4.04E-04	2.91E-03	1

APPENDIX C

PRECALCULATED N, A, CDFA VALUES FOR PRA-BASED COMPONENTS

APPENDIX C

PRECALCULATED N, A, CDFA VALUES FOR PRA-BASED COMPONENTS

TABLE C.1. Pre-Calculated N, A, and CDFA Values for PRA-Based Components

	Risk Importance	A (per hr per yr)	CDFA (per yr cubed)
11. Diesel	9.2E-4	1.6E-6	1.3E-5
16. a. Thermostat	6.0E-3	1.5E-7	7.9E-6
b. XFER switch	4.7E-6	2.3E-7	9.5E-9
c. Bistable	1.2E-5	1.4E-7	1.5E-8
17. a. Relay	4.8E-2	9.1E-8	3.8E-5
b. Breaker	7.2E-2	1.6E-8	1.0E-5
18. a. A.O. valve	3.2E-2	4.0E-7	1.1E-4
b. Check valve	1.8E-2	3.8E-7	6.E-5
c. Hydraulic valve	3.3E-4	6.5E-6	1.9E-5
d. Manual valve	3.3E-2	2.2E-8	6.4E-6
e. M.O. valve	2.2E-2	2.6E-7	5.0E-5
f. S/R valve	3.5E-4	6.7E-7	2.1E-6
19. a. Motor pump	6.7E-3	2.2E-7	1.3E-5
b. Turbine pump	9.3E-3	2.7E-6	2.2E-4
21. Turbine	1.0E-4	3.7E-6	3.2E-6
22. Heat exchanger	4.4E-3	1.4E-7	7.8E-6
23. Air conditioner	7.7E-4	5.6E-7	3.8E-6
24. Fan	7.6E-3	2.1E-7	1.4E-5
25. Battery	9.2E-4	3.4E-8	2.7E-7
26. a. Battery charger	1.1E-4	3.5E-8	3.4E-8
b. Inverter	4.2E-4	4.9E-6	1.8E-5
c. Rectifier	4.7E-4	8.7E-8	3.6E-7
27. Transformer	1.2E-2	1.7E-8	1.8E-6
29. Tank	2.7E-2	2.7E-11	6.4E-9
30. a. AC bus	4.3E-2	1.1E-11	4.1E-9
b. DC bus	1.1E-1	1.1E-10	1.1E-7

APPENDIX D

DETAILED RATIONALE FOR PANEL'S FINDINGS FOR PRIORITIZATION USING RISK-BASED CRITERIA

APPENDIX D

DETAILED RATIONALE FOR PANEL'S FINDINGS FOR PRIORITIZATION USING RISK-BASED CRITERIA

The enclosed rationale are presented line by line for the S/Cs from the data presented in Table 2.7. The S/Cs are arranged by their TIRGALEX S/C numbers (e.g., 1:RPV, etc.), not by their order in the table.

1: Reactor Pressure Vessel (RPV)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-0	2.0E-12	120	0.90	0.10	1.4E-5	3

N: Failure in the RPV (defined as gross rupture) results directly in core damage because safety inspection systems are not capable of providing sufficient core cooling.

A: The panel requested a specific notation of these opinions that the model does not address pressure vessel aging well because the vessel lifetime is so long. There are no documented failures, so the age factor was considered low.

L_{ind}: 120 months. This value was based on the ten-year in-service inspection frequency for the pressure vessel per American Society of Mechanical Engineers (ASME) Section XI requirements. Surveillance coupons are also pulled and examined at this time.

P_D: 90% -- Based on well developed technology of nondestructive examination (NDE) inspection methodology.

$P_{R/D}$: 10% -- This low value reflects the fact that a pressure vessel has not, to date, ever been successfully repaired (annealed or otherwise stress relieved in the beltline region). Mitigation is currently handled through flux leakage management programs aimed at aging rate reduction, not repair.

2.a: Containment (BWR-Mk-I)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-0	1.0E-7	18	0.90	0.80	1.9E-3	5

N: The "component" in this case is the BWR Mark I containment wetwell which was separated from other containment components for special considerations. However, all containment structures were given the highest risk importance.

A: Aging issues for the BWR Mark I containment wetwell were felt to be more severe than for other concrete structures due to wet well problems with corrosion.

L_{ind}: 12 months. Reflects visual inspection of torus and associated components.

P_D: 90% -- Corrosion and other age related manifestations are being given high priority by BWR Mark I owners.

$P_{R/D}$: 80% -- Usually radiographic documentation or other clearly discernable correction.

2.b: Containment (other)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-0	1.0E-13	60	0.10	0.50	7.0E-7	2

N: The importance for all containment and concrete structures was set to the highest level of risk significance as discussed in Section 3.

A: The aging factor was considered to be low for the majority of the concrete structures since they are designed to withstand environmental and stress conditions.

L_{ind}: 480 months. Reflects the panel opinion that the containment is basically uninspected following construction.

P_D: 10% -- Age related information is not specifically addressed by current surveillances.

$P_{R/D}$: 50% -- The failure (defect) observed may not be significant in terms of containment integrity.

3: Other Concrete Structures

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-0	1.0E-13	60	0.10	0.50	7.0E-7	2

N: Buildings and other concrete structures adjacent to equipment were of highest risk importance since they were felt to be of equal risk significance to the pressure vessel; e.g., failure (collapse) of a building well directly result in core damage due to damage to equipment severing of pipes and cables. Seismic events are the initiators.

A: The stipulation for the age factor was that it was to be the lowest on the ranking. It was recognized that aging may even strengthen the concrete.

L_{ind}: 480 months. This indicates that these structures are basically not effectively inspected during their nominal 40 year design life.

P_D: 10% -- Reflects only exterior visual inspection.

$P_{R/D}$: 50% -- Identified failure (deficiency) may not be significant to containment integrity.

4.a: Reactor Coolant Piping and
Safe Ends (Large LOCA)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-4	1.0E-12	0	0	0	n/a	1

N: For 20" pipe, a large break was assumed to be beyond the makeup capability of HPCI. The conditional probability for core damage given large LOCA is generally set in PRAs at 1.0E-4.

A: The small LOCA PWR category was set at 3E-11, and the large LOCA was considered still less likely than that.

(Eliminated from further consideration by screening criterion (ΔR_C <1E-7 CD/yr.)

4b: PWR Pipe (Small LOCA)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-3	1.0E-12	0	0	0	n/a	1

N: A small break LOCA of a pipe in the size of 20" in a PWR was considered for this grouping, a small break being one such that HPCI could keep up with it.

A: The aging factor for the PWR small LOCA was considered 3 orders of magnitude lower than that for a BWR since BWR piping exhibits higher stress corrosion rates than does PWR piping.

(Eliminated from further consideration by screening criterion (ΔR_C <1E-7 CD/yr.)

4c: BWR Pipe (Small LOCA)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-3	3.0E-8	36	0.20	0.90	3.7E-5	3

N: The risk importance of 20" pipe with a small break such that HPCI can keep up with it was felt to have the same conditional aspects as other small LOCAs, and was therefore set at 1E-4.

A: BWR stress corrosion was seen as the primary aging mechanisms so the following formula used to obtain the aging rate: 1E-03 (which is small LOCA-BWR)/(10 yr. doubling time x 8000 hrs/yr) = 3E-08.

L_{ind}: Weighted average based on detection of >50% through-wall cracks.

P_D: No defined test for wall thinning currently established. Anchored on 50% through-wall => 90% detection probability.

$P_{R/D}$: Replacement is method of repair with assumption that radiography and post-repair hydro will be required.

5a: Large Other Safety Piping

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
6.4E-3	3.0E-9	18	0.10	.90	2.3E-5	3

N: Service water system is the surrogate system chosen; piping 10-24 in. This system is a vital link in preventing decay heat from causing core damage in a post-trip condition and in providing cooling to key safety equipment. Piping only is considered in this category.

A: The failure mode considered here was a complete failure of the piping network to deliver flow required to the component cooling and RHR heat exchangers.

L_{ind}: 18 months. This corresponds to flow surveillance testing of the system.

P_D: Aging mechanisms are not, in general, addressed by the system surveillance which basically requires only a minimum flow rate to be met in a given component branch.

$P_{R/D}$: Identified failures are corrected and tested for compliance.

5.b: Small Other Safety-Related Piping

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-3	3.0E-7	60	0.10	0.90	2.1E-3	5

N: Letdown and RWC piping (6-10 in.) are the surrogate. The dominant risk contributor in this category of pipe was considered to be a small LOCA. Thus, the risk importance is the same as for other small LOCAs. This scenario considered a small break such that HPCI can barely keep up with it; approximately a 1" break or larger.

A: Smaller pipe was addressed in this category, and the aging acceleration was felt to be one order of magnitude larger than the BWR-Small LOCA case for RCP pipe, namely 3E-7.

L_{ind}: 60 months. This represents a system weighted average aimed at crack detection.

P_D: 10% -- Wall thinning is not considered. Crack resolution is assumed at 90% confidence level for 50% through wall or more.

$P_{R/D}$: 90% -- Detected problems are weld repaired and confirmed by radiography.

6.a: Steam Generator (S/G) Tubes

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
3.0E-4	5.0E-6	36	0.50	0.50	9.5E-4	5

N: It was assumed that the concern was tube rupture in the primary to secondary system connection. N was felt to be moderate because there are mitigating steps available and reasonably well-defined operator actions for mitigation depending on the scenario and plant-specific conditions. This event was seen as more likely to lead to core damage than a small LOCA; an importance of 3E4 was assigned.

A: For a tube leak between the primary and secondary sides, many failure mechanisms exist and all were felt to be attributable to aging. A S/G tube leak was estimated to occur every year; its age factor was considered high.

L_{ind}: 36 months. This value reflects the rotational inspection of a representative population (typically 10%) and assumes that failures in the tested population will trigger increased inspection.

P_D: 50% -- Age detection methods are currently not reliable. This figure obviously related only to the population that is inspected.

$P_{R/D}$: 50% -- Definition is sensitive here. Repair is by the affected tube plugging and does not return the component to new condition. The steam generator is over designed such that, up to a point, plugging does not degrade the S/G's ability to perform its safety function.

6b: Steam Generator Shell

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-5	1.0E-12	0	0	0	n/a	1

N: The risk importance of this category was considered equivalent to a main steamline break inside containment.

A: The aging factor was set at 1 order of magnitude larger than a small break/large pipe in a PWR.

Eliminated from further consideration by screening criterion (ΔR_C <1E-7 CD/yr.)

7: Reactor Coolant Pump Casing

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-4	1.0E-12	0	0	0	n/a	1

N: Same risk importance assumed as for large break LOCA.

A: The wall thickness for the casing is twice that for pipe and the ASME code requires inspection of pump casings. No problems have been found. Therefore, the aging factor was felt to be extremely low.

Eliminated from further considerabion by the screening criterion (ΔR_C <1E-7 CD/yr.)

8: Pressurizer

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-4	1.0E-12	10	0.90	0.90	1.2E-12	1

N: One of the panel members had experience with a heater weld leak and felt that the risk importance should be the same as a small break LOCA. It was also observed that upon failure, the pressurizer cannot be isolated.

A: Since these are all clad pieces and vessels, corrosion was seen as a second or third order effect. This would be in the same category as the pressure vessel or reactor coolant piping and safe ends for a large break LOCA.

L_{ind}: Averages in ASME Section XI and continuous monitoring via leak detection.

P_D: Detection mechanism is visual inspection, NDE methods, and leaks. Frequency number is a composite of these.

$P_{R/D}$: Any failure (leak) represents a small break LOCA so repair is assured.

Post-workshop review result in component eliminated by screening criterion (R_C <1E-7 CD/yr.)

9a: Control Rod Drive Mechanisms (BWR)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-1	3.0E-9	18	0.90	0.90	4.5E-6	2

N: Standby liquid control is the only available backup assuming that the CRDM has failed. 0.1 was considered a reasonable rate for the SLC system and the required operator action.

A: The CRDMs are known to be subject to aging and there has been leaking. The aging rate was felt to be worse than that for BWR pipe due to the smaller diameter and the stress level involved. It should be noted that the aging factor does not account for the solenoid valves, which were felt to be the dominant failure mechanism.

L_{ind}: Follows Tech Spec guidelines.

P_D: Weighted for Tech Spec initiated test criteria failures and resulting replacement of piece parts.

$P_{R/D}$: Influenced by rigorous surveillance and post-maintenance test requirements. Normal practice is to remove, rebuild or replace CRDMs each refueling outage such that in ten years all drives will have been completely replaced, setting the clock to zero.

9b: Control Rod Drive Mechanisms (PWR)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-3	3.0E-11	18	0.50	0.90	1.5E-9	1

N: The risk importnce for these was considered a factor of 20 to 30 lower than the BWR CRDM because ATWS mitigation is easier (primary, secondary cooling).

A: The PWR CRDM aging factor was felt to be two orders of magnitude better than for the BWRs because of the multiplicity and design. With the BWR solenoid, one train is kept energized and is therefore more susceptible to aging.

L_{ind}: Weighted for Tech Spec requirements to replace any component pieces that indicate degradation through surveillance testing.

P_D: Rigorous testing, but not all pieces reveal degradation in tests now administered.

$P_{R/D}$: Stringent post-maintenance test requirements assumed on this component.

10.a: Cables

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.1E-1	2.7E-9	60	0.10	0.90	2.1E-3	5

N: Risk importance was assumed to be similar to that of a DC bus. This is based upon the assumption that there exists at least one DC cable for each DC bus whose failure has the same effect as the failure of the bus. These are the risk-dominant cables.

A: Considering the top 10% of the cables in terms of importance, the aging factor was treated as nearly the same as that for a transformer, or 2.7E-9.

L_{ind}: 60 months; no systematic inspections. A series of failures of a particular type would trigger an investigation of that type and manufacture only.

P_D: 10% -- Shear volume, diversity of types and the virtual impossibility of complete surveillance was the limiting consideration.

$P_{R/D}$: 90% -- Once detected, the accepted repair procedure is to replace the cable and with high probability on all similar applications. This historically has a high rate of success. See comments on connectors.

10.b: Connectors

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
2.0E-2	2.7E-8	60	0.20	0.90	1.8E-3	5

N: The panel judged connections at a lower risk importance relative to cables since they are used in fewer applications than cables. The circuits often have diverse channels for functional aspects, and power cables often don't use connectors. This is based on the assumption that there does not exist one connector whose failure is equivalent to failure of the bus (see cables).

A: The aging acceleration was considered by the expert panel to be one order of magnitude worse than the cables, or 2.7E-08.

L_{ind}: 60 months. This figure is based upon the fact that there is no systematic test or inspections but that a series of failures of a specific connector may be viewed as a generic indication which would trigger increased surveillance of like items with possible wholesale replacement.

P_D: 20% -- This relatively low value reflects the fact that there is a sizeable population of diverse types and the lack of age related investigation of these various types. Orientation is aimed at failures.

$P_{R/D}$: 90% -- Repair probability is based on the diversity of the component and endeavors. The lack of a systematic effort to investigate root causes and carry out a full-scale investigation of all types regardlessof failure exposures.

11: Diesel Generators

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
2.0E-2	3.6E-6	3	0.30	0.90	2.7E-4	3

N: Recent PRAs have found loss of offsite power as among the highest contributors to risk whereas the 4 PRAs considered in this study do not accurately capture the LOSP risk.

A: The aging rate is high and the primary reason is believed to be testing, especially fast-start testing. Research is needed in this area and should continue to be focused on the large body of diesel information (nuclear and non-nuclear) available to uncover other aging causes within and outside testing and to document them so action can be taken.

L_{ind}: 3 months. This is based on major tests which include load sequencing of the vital bus. This is a weighted figure which integrates the complex multi-system nature of the emergency diesel.

P_D: 30% -- This is an integrated value which recognized various sub-system data frequencies.

$P_{R/D}$: 90% -- This value acknowledges post-maintenance surveillance testing.

12: Reactor Internals

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-1	2.0E-9	18	0.10	0.90	2.4E-4	4

N: For a significant but partial coolant blockage, the importance was considered less than that for the reactor vessel but greater than that for ATWS (but same as ATWS for BWRs).

A: Stainless steel swelling and stress corrosion cracking do occur and partial support failure requires less severe conditions than the RPV so the age factor was established at an order of magnitude greater than the reactor pressure vessel.

L_{ind}: 18 months. This is based on inspection during refueling outage coupled with on-line instrumentation of vessel condition (principally ΔP).

P_D: Difficult to examine at an age detection level -- few existing surveillance parameters which reflect aging mechanisms.

$P_{R/D}$: Once detected, a failure will require the reactor to be shut down until repairs are completed and testing is satisfactory.

13: PWR Pressure Vessel Support

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-7	1.0E-12	0			n/a	1

N: The definition of this category was PWR sliding foot. If the sliding foot is removed, it was believed that the vessel would be supported by the pipes, since steam generators have been known to be held up this way.

A: Not seen as an issue.

Eliminated from further evaluation by this screening criteria (ΔR_C <1E-7 CD/yr.)

15: Snubbers

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.1E-6	5.1E-6	18	0.90	0.90	8.4E-8	1

N: Snubbers were not considered to play a risk-significant role except for those on the steam generators under seismic conditions. The risk importance was felt to be around the same level as for the turbine.

A: Aging considerations were high since the snubbers won't do their job if they are not maintained, and the aging rate was seen to be similar to that for bolts.

L_{ind}: Accounts for visual walkdown outside containment, and assumes that surveillance failures lead to a higher inspection frequency.

P_D: Estimate may be somewhat high due to lack of standardization in testing, but assumption that failures trigger an increase in surveillance scope still makes it reasonable.

$P_{R/D}$: Extensive post-maintenance testing ensures successful repair to the extent possible.

16a: Thermostat

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
6.0E-3	1.5E-7	18	0.7	0.9	2.2E-5	3

N: This is a subgroup of Instruments and Controls (I&C), which were considered to be highly correlated but diverse. The generic 4-plant PRA value was deemed to be satisfactory.

A: Considerations of equipment qualifications since TMI; field monitoring veruss control environment, which should be considered separately, and doubling times suggested that the precalculated values were satisfactory (unless field mounted and control environment I&C would later be considered separately).

L_{ind}: 18 mo

P_D: 70%

$P_{R/D}$: 90%

16b: Transfer Switch

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
4.7E-6	2.3E-7	18	0.2	0.9	3.3E-7	1

N: A subgroup of I&C which are highly correlated but diverse. The 4-plant PRA value was deemed satisfactory.

A: Same rationale as for 16a. Thermostat.

L_{ind}: 18 mo

P_D: 20%

$P_{R/D}$: 90%

16c: Bistable

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.2E-5	1.4E-7	18	0.7	0.9	4.2E-8	1

N: A subgroup of I&C, which are highly correlated but diverse. The precalculated value was deemed satisfactory.

A: Same rationale as for Thermostat (16b)

L_{ind}: 18 mo

P_D: 70%

$P_{R/D}$: 90%

17.a: Relay

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
4.8E-2	2.5E-7	6	0.20	0.90	4.0E-4	4

N: Relays were considered vulnerable to common mode failure, especially in a seismic event, and due, to their involvement in critical paths for nearly all functions, they were seen as highly risk significant.

A: Relays experience degradation due to heat and humidity and are more susceptible to seismic common mode failures when in an aged condition. Contact degradation (pitting and arcing) also occurs as an age related failure mechanism.

L_{ind}: 6 months. This is a weighted estimate for all safety significant relays based on the periodic testing, combined with system response-time testing done once every refueling on key protection systems.

P_D: 20% -- Reflects the basic lack of aging determination in this component until substantial numbers of failures have been experienced, and may also be limited to a particular type and manufacture. Repair is usually accomplished by replacement.

$P_{R/D}$: 90% -- Shows the above weighting of differing relay types and manufacture, and repair by replacement techniques.

17b: Breaker

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
7.2E-2	1.6E-8	18	0.50	0.90	5.6E-5	3

N: Risk importance of breakers was considered high due to their fire sensitivity, lack of backup, and far-reaching impact on many critical components should they fail. Int his case, it was assumed that there was more than one breaker associated with each major AC bus whose failure would have the same effect as failure of the bus. Thus, the risk importance of breakers as a group would be somewhat higher than AC buses as a group.

A: Aging mechanisms are associated with contact burnout and the close/open coils, but the modes are detectable. Breakers were considered accessible, maintainable, and a static technology.

L_{ind}: A weighted average based on quantity and diversity of type and manufactures of safety related breakers.

P_D: High incidence of breaker failure connotes possible undetected aging phenomena.

$P_{R/D}$: Reflects long historical experience (nuclear and non-nuclear) and state of technology.

18a: Air Operated Valves (AOV)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
3.2E-4	4.0E-7	18	0.50	0.90	6.2E-6	2

N: Redundancy exists to back these up, in the form of MOVs, other AOVs, etc. AOVs are in one of three or four systems available to mitigate core damage, so the importance rank reflects this.

A: AOVs are in critical functions, and their air lines, due to their size, have failure rates of 1E-01 to 1E-02 per year, so the age factor was set to reflect these considerations.

L_{ind}: 18 months. Normal satisfactory operation or partial stroke tests are not considered here.

P_D: 50%--Based on surveillance testing during shutdown (disassembly and inspection as well as functional checks).

$P_{R/D}$: 90%--Due to the safety importance of these valves thorough post-maintenance testing.

18b: Check Valves

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
8.0E-4	3.8E-9	18	0.10	0.90	3.7E-6	2

N: Two failure mechanisms were considered: 1) extremely high back-pressure from, for example, the reactor coolant system to a low pressure system such that a safety valve will not relieve the pressure, and 2) failure to open or close upon demand (or disk falling off, which would produce same result). The risk importance of the first mechanism is high and the second low, so the selected importance is based only on the first mechanism.

A: The aging influences involved in the failure mechanisms cited above were considered lower for the first item and higher for the second; the aging rate balanced out to a median level.

L_{ind}: This is a composite figure which covers all safety-related check valves including testable checks.

P_D: Back leakage gives highest indication of aging problems.

$P_{R/D}$: Best you can attain because of common mode failure. Assumes complete post-maintenance testing.

18c: Hydraulic Valves

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-5	1.3E-7	18	0.50	0.90	6.3E-8	1

N: These valves are not used in a BWR for feedwater or main steam isolation and the PWR closing rate requirements are not risk-significant.

A: The NPRDS data base used for the initial ranking represents a small population and a relatively high number (2) of failures so the original ranking was seen as being artificially high. The hydraulic actuators have a proven history, but the accessories that go with them may affect the aging factor. These valves are used to protect the turbine and generator from runaway and should be considered to be reliable since there is a significant experience base. If they were not reliable, there would be two valves in place rather than one, because when one valve fails the turbine is lost.

L_{ind}: No credit is given for partial stroke tests, so only major surveillance intervals are counted.

P_D: Current surveillances give little aging information per se. Failure of stroke time testing generally results in valve dissassembly which reveals the valve's age status.

$P_{R/D}$: Post-maintenance testing gives a clear picture of repair success.

18d: Manual Valve

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-5	2.2E-9	60	0.50	0.60	2.7E-8	1

N: The present ranking reflects administrative failure (e.g. operator error in locking valves shut), not actual valve internal failure. Also, 162 valves were modeled in the original data set so the assumptions made are multiplied. Still, the the failure mechanisms are not believed to significantly affect risk.

A: Although these valves are subject to aging mechanisms such as erosion/corrosion and stem breakage, there was not sufficient impetus to warrant changing the ranking.

L_{ind}: Incorporates Tech Spec requirements, operational checks, and utility initiatives.

P_D: Includes all safety-related valves, and represents current inspectionm ethods.

$P_{R/D}$: Based on evidence of recurring problems.

18e: Motor Operated Valves (MOV)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
2.2E-2	3.6E-6	3.9	0.70	0.90	5.6E-5	3

N: The MOV risk importances vary over many orders of magnitude; it was decided to assign a moderate importance to reflect this.

A: There are as many failure mechanisms for MOVS as there are for other valve types plus additional failure mechanisms associated with the valve operator and intermittent duty cycle that are omnipresent and key to MOVs.

L_{ind}: 3 months. This is a composite figure of all safety related MOVs based on stroke testing.

P_D: 70%--Use of systems like MOVATS during stroke testing is significantly increasing aging detection.

$P_{R/D}$: 90%--Valves must meet stringent post-maintenance stroke testing requirements.

18.f: Safety/Relief Valve (S/RV)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-4	6.7E-7	18	0.90	0.90	1.0E-6	2

N: In BWRs, two S/RVs sticking open have caused a LOCA in the past. RCIC can compensate for one stuck open S/RV, but not two. Therefore, the risk importance rank is greater than three, but not as high as five (on the relative scale).

A: The expert panel felt that S/RVs do age so the ranking in this category should be high. The "saving grace" was considered the maintainability accounted for in the P values of the model.

L_{ind}: 90% of valves every 18 months. Tech Spec regulated.

P_D: Very high due to sophisticated equipment which is available to test the actuation, blowdown, and closure specifications as a result of previous safety concern resolution.

$P_{R/D}$: High due to extensive operating and test experience.

19a: Motor Driven Pump

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
6.7E-3	2.2E-7	18	0.70	0.90	2.2E-5	3

N: Despite the fact that MD pumps are in both mitigation and support systems with no backup, when compared with the turbine-driven pumps and when determined that station blackout AFW and RCIC needs drove the turbine pump importance up, it was decided to leave the MD pump rank at a medium level.

A: Motor driven pumps were considered relatively simple in nature and a long experience base exists in comparison to the turbine pumps. The aging rate was not felt to be as high. Aging effects exist none the less due to the surrounding environment, thus keeping the MD pumps relatively high in ranking.

L_{ind}: A weighted average of safety related units which considers various types of surveillances.

P_D: A wide range of opinion which seemed to be based on variations in experiences. The midpoint seemed appropriate.

$P_{R/D}$: Fairly well defined repair practices and tests following repair.

19.b: Turbine Driven (TD) Pump

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
9.3E-3	2.7E-6	12	0.50	0.90	5.4E-4	4

N: Station blackout-related needs for AFW (PWR) and RCIC (BWR) were considered the driving factors for the relatively high TD Pump risk importance.

A: The turbine driven pump was considered more prone to aging-related degradation due to the governor control system and its function as a "water wheel" in the AFW and RCIC systems. It is therefore more susceptible to mechanical and auxiliary system damage (than the motor driven pumps, for example) and merits the high ranking.

L_{ind}: 12 months. This reflects a weighted factoring of full surveillance texts (full flow) with more frequent, but less in-depth surveillance information.

P_D: 50% -- Factors include predictive testing (oil analysis, vibration measurement, etc.) with other surveillance information.

$P_{R/D}$: 90% -- Complete repair is assumed for these components with post-maintenance surveillance testing.

20: Motors

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
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N: Motors were considered under their specific motor operated equipment groups such as motor operated valves and motor driven pumps.

21: Turbine

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-6	1.0E-7	60	0.90	0.50	5.4E-8	1

N: Turbine failure was considered separately from turbine missiles. The original turbine ranking was based on turbine driven pumps. Additionally, though turbines fail regularly, the risk importance of the failures was still felt to be low. Even in the case of turbine missiles, they would have to strike the core directly or go into the control room to cause core damage.

A: The original ranking reflected a data set which implied a failure (not trip) of every turbine each year, and this seemed unrealistically high. Turbine problems were seen to be centered in a couple of poorly-monitored systems and poorly-constructed turbines. The consideration of this category as being the turbine generator was not agreed to by the group due to the introduction of susceptability of stray currents rather than mechanical aging mechanisms.

L_{ind}: Basically, this represents a tear down frequency to determine degradation.

P_D: Complete examination for degradation is assumed.

$P_{R/D}$: Even with post-maintenance surveillance testing, experience with turbines has been similar to diesels, not real great.

22: Heat Exchangers

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
6.4E-3	1.4E-8	3	0.10	0.90	3.0E-6	2

N: The high risk importance is due to the assumed loss of heat exchanger function for a key heat exchanger (like CCW) leading to loss of one train of nearly all safety systems, and loss of room cooling for one train of RHR, HPCI, and RCIC; it was assume that one train of RHR was available.

A: Cracking, flaws, and leaks are failures that compromise the HX function. Due to the 22 year doubling time of the failure rate, the aging factor was considered to be on the low side.

L_{ind}: Composite number that averages operational use with surveillance testing.

P_D: Most tests do not directly measure the heat transfer capability or extent of corrosion of the unit.

$P_{R/D}$: Complete restoration would be proven through post-maintenance testing.

23: Compressor (Instr. Air)

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
5.0E-4	5.0E-7	6	0.20	0.90	8.4E-6	3

N: These include service and instrument air compressors, but not diesel air compressors. These were considered to have a medium risk importance because: (a) older plants usually have a non-safety-related design and (b) the loss of instrument air can cause many transients not provided for in system design.

A: Oil is the predominant failure mechanism for compressors and it is aging-related due to seal failure so the ranking was seen as being relatively high. NOTE: Compressors include dryers as part and parcel of the system and consider the aging-related failure mechanism of rust.

L_{ind}: 6 months. Complete functional checks.

P_D: 20%--Aging principally detected through oil analysis and vibration measurements. Aging usually found when correcting a failure.

$P_{R/D}$: 90%--Assumes confirmation through post-maintenance testing.

24a: Chillers

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
6.0E-4	1.5E-6	18	0.70	0.90	2.2E-5	3

N: Risk importance for fans and chillers were considered equivalent. These essential equipment are considered as room coolers; they were felt to impact containment integrity only when all essential equipment except for room coolers has failed. Also, it was agreed that the risk importances should be lower than those of heat exchangers and batteries and about equivalent to air compressors.

A: The aging rate was felt to be three-quarters of an order of magnitude higher (worse) than that of the motor-driven pumps.

L_{ind}: 18 months. Nominal major preventive maintenance frequency for this component.

P_D: 70%--Considerable aging evaluation of the standby unit is performed.

$P_{R/D}$: 90%--Based on extensive post-maintenance surveillance testing.

24b: Fans

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
6.0E-4	2.1E-7	18	0.70	0.90	3.1E-6	2

N: Risk importances for fans and chillers were considered equivalent. As room coolers, they were felt to impact containment integrity only when all essential equipment except the room coolers had failed.

A: The fans themselves age due to corrosion and the fan motors also experience aging. This combined with a five-year failure rate doubling time contributed to the aging rate determination.

L_{ind}: Based on surveillance requirements and failure indicators (no flow indication on effected piece of equipment).

P_D: Aging leads to degraded performance which will be detected by surveillance testing.

$P_{R/D}$: Performance testing following repair.

25: Battery

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
2.0E-2	3.4E-7	6	0.90	0.90	1.1E-5	3

N: The consideration involved just the batteries themselves and it was felt that the risk importance should be in the same range as the diesel generator (3 to 4) (because of similar roles in station blackout); so a high 3 rank was agreed upon.

A: The panel agreed that batteries are susceptible to aging failure mechanisms so the ranking had to be rather high. One panel member pointed out that the doubling time would work out to a number of months if the rank was set to 5, and that was considered unrealistic (as was the 29-year doubling time given as the original data set value). It was thus agreed that a rank of 4 for aging acceleration, corresponding to a doubling time of 3 years, was most appropriate.

L_{ind}: This is a composite based on frequent non-aging specific daily checks coupled with longer intervals involving more in-depth testing.

P_D: Recognizes the recent increases in testing for aging deterioration on this component.

$P_{R/D}$: Repair is usually by replacement, followed by testing.

26.a: Battery Charger

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.1E-4	3.5E-8	12	0.90	0.90	2.6E-8	1

N: The batteries, chargers, inverters, and rectifiers were compared in risk importance to aid in their ranking. Chargers, then, were considered less risk significant than batteries and were left at the original ranking.

A: No impetus to change original ranking.

L_{ind}: This figure is a weighted average of all monitoring incuding daily checks through full surveillance load testing during refuel outages.

P_D: Due to attention this device has received, aging detection is considered to be very complete.

$P_{R/D}$: High post-maintenance testing requirements ensure complete repair.

26b: Inverter

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
4.7E-6	4.9E-6	12	0.50	0.90	5.0E-7	1

N: The importance to risk of the inverter was debated heavily between a preference for a lower ranking based on the greater importance of the DC bus in the ability to change equipment states, and the contention that the AC bus provides plant status monitoring through instrumentation and is therefore essential to evacuation planning. A decision was ultimately made to retain a median ranking.

A: Age-related degradation is a factor due to the inside-the-cabinet environment, and this justifies the high rank.

L_{ind}: Same as battery chargers.

P_D: Aging phenomena are difficult to detect, resulting in a medium detection probability.

$P_{R/D}$: Once detected, part replacement and post-maintenance testing provide an excellent probability of repair.

26c: Rectifier

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
4.7E-6	8.7E-8	12	0.50	0.90	8.9E-9	1

N: The rectifier risk importance was generally agreed as being the same as that for inverters and was thus set at a median rank.

A: Considered to be about the same as that for the battery chargers.

L_{ind}: Same as inverters.

P_D: Same as inverters.

$P_{R/D}$: Same as inverters.

27: Transformer

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.2E-2	1.7E-9	18	0.70	0.90	5.1E-7	1

N: The 4160 volt and offsite power transformers were considered in the ranking. The risk importance was seen to be high since a transformer loss leads to multiple simultaneous equipment losses, similar to, but not as serious as, loss of an AC bus (since backup power can usually be provided by diesels or crosstie breakers).

A: Industrial experience shows that transformers have a constant failure rate over many years of service, so age was not considered to be a significant factor in transformer failure.

L_{ind}: Gives some weight to ability of frequent (daily) checks to detect trends, coupled with long term oil analysis.

P_D: Environmental variations usually mask temperature variations as an aging indicator. Oil analysis is most reliable.

$P_{R/D}$: Very high due to equipment importance and post-repair surveillance requirements.

28: Fuel Storage Rack

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-6	5.0E-12	--	--	--	--	1

N: Risk importance was considered low for the fuel rack, and at an equivalent level to the turbine and snubber risk significance.

A: A passive component constantly immersed in water, the fuel racks were seen to be affected by aging similarly to the RPV or the turbine.

Eliminated from further consideration by screening criteria (ΔR_C <1E-7 CD/yr.)

29a: Medium Pressure Tank

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
6.0E-3	1.0E-12	12	0.10	0.70	5.4E-9	1

N: These were seen as tanks in the 50 to 100 psi range and were felt to be very important since they are generally surge tanks and boron injection tanks. Losing one was considered equivalent to losing a train of an important support system.

A: Leaks occurred but not failures; assumed they were designed to withstand contents such as boron, so the aging rank is higher than for accumulators, but not by much.

L_{ind}: Based on Tech Spec visual inspection requirements.

P_D: Low because of inspection methods.
Until you get a leak, nothing happens.
Lagging on some.

$P_{R/D}$: Good based on post-repair testing.

29b: Atmospheric Pressure Tank

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
2.5E-2	2.0E-10	12	0.10	0.70	2.2E-6	1

N: Considered important to risk since the best examples of these tanks is the Refueling Water Storage Tank and the Condensate Storage Tank, which supplies water to many safety-related systems.

A: Only pinhole leaks were anticipated over time, so aging failure is not a predominant failure issue.

L_{ind}: Reflects Tech Spec compliance.

P_D: Only visual checks are performed on these tanks. Exterior lagging may hinder identification of leakage.

$P_{R/D}$: Once detected, repair of this and any additional leakage is highly probable.

29.c: High Pressure Tank

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-4	1.0E-12	0	--	--	--	1

N: The original category of accumulators/tanks was subdivided on the basis of the pressurization levels. The accumulator/high pressure tank category was seen to have a risk importance at a low 2 rank since a total functional failure (depressurization or loss of valves between tank and vessel) of one tank would still leave others available for backup.

A: Experience extends to nearly 200 reactor years with few problems; aging is not believed to be a significant factor.

Eliminated from further consideration by the screening criterion (ΔR_C <1E-7 CD/yr).

30a: AC Bus

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
4.3E-2	1.1E-9	18	0.50	0.90	2.3E-6	2

N: The AC bus was felt to have a high risk significance due to its support of plant status instrumentation and to the fact that its failure could lead to widespread Instrument and Control (I&C) disablement. Still, the risk importance was considered half an order of magnitude lower than the DC bus.

A: The age factors were considered the same for AC and DC buses and are discribed under the DC bus (30.6).

L_{ind}: Critical buses can only be inspected in a de-energized condition, i.e., during shutdowns.

P_D: Specific bus deterioration tests are not currently established.

$P_{R/D}$: Repair is generally by replacement, and follow-up testing is assumed.

30b: DC Bus

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.1E-1	1.1E-9	18	0.50	0.90	5.9E-6	2

N: The DC bus was defined as the wiring that goes into the distribution box and breakers. Due to the effects of the bus on a multitude of control and command units, the risk importance was seen to be high.

A: Equipment qualification testing showed bus failures due to high humidity, steam, and chemical factors but these failures were the same whether the bus was one or thirty years old, so these mechanisms aren't aging related. The failure mechanisms which are age-dependent are dust buildup to connective paths (causing shorts) and how well the terminal block screws retain the torque level.

L_{ind}: 18 months. This bus must be de-energized to perform surveillances, hence they correspond to the refueling shutdown interval.

P_D: 50%--Based on current detection methods.

$P_{R/D}$: Post-maintenance surveillance.

31: Bolts

N	A	L_{ind}	P_D	$P_{R/D}$	RISK	FINAL RANK
1.0E-4	5.1E-7	18	0.50	0.90	2.5E-6	2

N: Bolts were considered important to risk due to their widespread existence in bracing primary and secondary side components such as valve bonnets, pumps, and steam generators. The potential for the "zipper effect" of bolt failure was also cited as an issue. The risk significance was assumed equivalent to a small break LOCA, since that seems to be the most likely risk-significant event resulting from bolt failure.

A: The aging contribution to bolt degradation was considered significant due to steam cutting and wastage (erosion from chemicals such as boric acid) and rusting.

L_{ind}: Assumes inspection on disassembled, safety-related, bolted components.

P_D: Depends on perceptible crack detection. Detection of a problem is assumed to result in extended surveillance of similar fasteners.

$P_{R/D}$: Flaw detection would result in fastener replacement.

APPENDIX E

EVALUATION OF AGING RESEARCH BENEFITS TO GSI RESOLUTION

APPENDIX E

EVALUATION OF AGING RESEARCH BENEFITS TO GSI RESOLUTION

TABLE E.1. Example of Format for Presentation of Prescreened GSIs

GSI A-11: REACTOR VESSEL MATERIAL TOUGHNESS

DESCRIPTION:

Decrease in vessel fracture toughness with accumulated neutron irradiation reduces safety margins.

STATUS:

Resolved, but work is continuing.

SAFETY ISSUE SCHEDULED RESOLUTION DATE:

RESOLUTION OF ESSENTIAL COMPONENT AGING TECHNICAL SAFETY ISSUES:

-RPV
-
-
-

COMMENT:

Resolved in October 1982. More work is needed in areas such as vessel annealing.

AGING RESEARCH IMPACT ON GSI: **YES**

TABLE E.2. Generic Safety Issues Scheduled for Resolution[(a)]

Issue Number	Title	Office/ Division/ Branch	Priority	Draft Resolution	ACRS Review	CRGR Review	Commission Review	Current Resolution Date	Prioritization Date
A.3,4,5	Steam generator tube integrity	NRR/DEST/EMTB	USI	04/83C	10/83C	10/83C	09/84C	12/87[(b)]	01/78
A.17	Systems interaction	RES/DE/EIB	USI	03/88	08/88	08/88	N/A	(04/89)	01/78
A.40	Seismic design criteria	RES/DES/EIB	USI	04/88	09/88	09/88	N/A	04/89	01/79
A.44	Station blackout	RES/DRPS/RPSI	USI	04/87C	05/87C	05/87C	11/87	(03/88)	01/79
A.45	Shutdown decay heat removal requirements	RES/DRPS/RPSI	USI	10/87	12/87	02/88	04/88	12/89	12/80
A.47	Safety implications of control systems	RES/DE/EIB	USI	01/88	07/88	07/88	N/A	04/89	12/80
A.48	Hydrogen control measures and effects of hydrogen burns on safety equipment	RES/DRAA/SAIB	USI	12/87	01/88	N/A	01/88	01/88	12/80
23	Reactor coolant pump seal failures	RES/DE/EIB	H	01/89	05/89	05/89	N/A	(04/90)	04/83
29	Bolting degradation or failures in nuclear power plants	RES/DE/EIB	H	11/88	03/89	05/89	N/A	10/89	08/82
77	Flooding of safety equipment compartments by back-flow through floor drains	RES/DE/EIB	H	(To be integrated into A.17)			12/88	09/83	
87	Failure of HPCI steam line without isolation	RES/DRPS/RPSI	H	04/89	07/89	07/89	N/A	(12/90)	09/85
93	Steam binding of auxiliary feedwater pumps	RES/DRPS/RPSI	H	(Complete except for Generic letter)				12/87	10/84
94	Additional low-temperature overpressure protection for light water reactors	RES/DRPS/RPSI	H	04/88	05/88	05/88	N/A	12/88	07/85
99	RCS/RHR suction line interlocks on PWR	RES/DRPS/RPSI	H	01/88	02/88	02/88	N/A	04/88	08/85

TABLE E.2. (contd)

Issue Number	Title	Office/ Division/ Branch	Priority	Draft Resolution	ACRS Review	CRGR Review	Commission Review	Current Resolution Date	Prioritization Date
101	BWR water level redundancy	RES/DE/EIB	H	02/89	12/89	12/89	N/A	03/90	05/85
105	Interfacing systems LOCA at BWRs	RES/DRPS/RPSI	H	01/88	04/88	04/88	N/A	12/88	06/85
115	Enhancement of the reliability of Westinghouse solid state protection system	RES/DRPS/RPSI	H	11/88	02/89	02/89	N/A	05/89	07/86
121	Hydrogen control for large, dry PWR containment	RES/DRA/RDB	H	Recommendations on need for rulemaking				02/88	09/85
122.2	Initiating Feed-and-Bleed	NRR/DEST/SRXB	H						
125.II.7	Reevaluate provision to automatically isolate feedwater from steam generator during a line break	RES/DRPS/RPSI	H	10/88	03/89	03/89	N/A	08/89	09/86
128	Electrical power reliability	RES/DE/EIB	H	12/88	09/89	09/89	N/A	12/89	11/86
113	Dynamic qualification of large bore hydraulic snubbers	RES/DE/EIB	H	08/89	05/90	06/91	N/A	09/91	03/85
130	Essential service water pump failures at multi-plant sites	RES/DRPS/RPSI	H	02/89	05/89	06/89	N/A	09/89	03/87
134	Rule on degree and experience requirements for senior operators	RES/DRA/RDB	H	10/88	04/89	04/89	07/87	09/89	07/86
B-56	Diesel reliability	RES/DRPS/RPSI	H	04/88	06/88	06/88	N/A	12/88	11/83
C-8	Main steam line isolation valve leakage control systems	RES/DRPS/RPSI	H	11/87	04/88	05/88	N/A	02/89	11/83

TABLE E.2. (contd)

Issue Number	Title	Office/ Division/ Branch	Priority	Draft Resolution	ACRS Review	CRGR Review	Commission Review	Current Resolution Date	Prioritization Date
I.A.4.2 (4)	Review simulators for conformance	NRR/DLPQ/LOLB	H	03/85C	12/85C	02/86C	04/86C	10/87	11/83
I.F.1	Expand QA list	NRR/DLPQ/LQAB	H	Issue on hold[a]				(TBD)	11/83
II.B.5	Behavior of severely damaged fuel	RES/DRAA/AEB	H	12/86C	A	N/A	SA	06/94	11/83
II.B.5 (2)	Behavior of core-melt	RES/DRAA/AEB	H	12/86C	A	N/A	SA	06/94	11/83
II.C.4	Reliability engineering	RES/DRPS/RHFB	H	05/86C	11/87	N/A	N/A	06/88	11/83
II.E.4.3	(Containment) integrity check	RES/DRPS/RPSI	H	12/87	N/A	N/A	N/A	02/88	11/83
II.H.2	Obtain technical data on the conditions inside the TMI-2 containment structure	RES/DRAA/AEB	H	09/90	N/A	N/A	N/A	03/91	11/83
HF1.1	Shift staffing	RES/DRPS/RHFB	H	11/84C	08/86C	02/85C	N/A	06/88	10/84
HF4.1	Inspection procedure for upgraded emergency operating procedures	RES/DRPS/RHFB	H	TI 2515/79 issued on 06/15/86				07/88[b]	10/84
HF4.4	Guidelines for upgrading other procedures	RES/DRPS/RHFB	H	Transfer to RES not completed				06/89	10/84
HF5.1	Local control stations	RES/DRPS/RHFB	H	07/89	10/89	02/90	08/90	12/90	10/84
HF5.2	Review criteria for human factors aspects of advanced controls and instrumentation	RES/DRPS/RHFB	H	07/89	10/89	02/90	08/90	12/90	10/84
HF8	Maintenance and surveillance program	NRR/DLPQ/LPEB	H	(Policy statement to be issued)				01/88	10/84
51	Proposed requirements for improving reliability of open cycle service water systems	RES/DE/EIB	M	01/89	05/89	06/89	N/A	03/90	06/83

TABLE E.2. (contd)

Issue Number	Title	Office/ Division/ Branch	Priority	Draft Resolution	ACRS Review	CRGR Review	Commission Review	Current Resolution Date	Prioritization Date
70	PORV and block valve reliability	RES/DE/EIB	M	02/88	05/88	06/88	07/88	10/88	05/84
79	Unanalyzed reactor vessel thermal stress during natural convection cooldown	RES/DE/EIB	M	02/88	N/A	N/A	N/A	07/88	07/83
82	Beyond design bases accidents in spent fuel pools	RES/DRPS/RPSI	M	04/88	05/88	05/88	N/A	12/88	12/83
A-29	Nuclear power plant design for the reduction of vulnerability to industrial sabotage	RES/DRPS/RPSI	M	06/89	N/A	N/A	N/A	08/89	11/83
B-5	Ductility of two way slabs and shells and buckling behavior of steel containments	RES/DE/EIB	M	03/88	08/88	08/88	N/A	03/89	11/83
B-17	Criteria for safety-related operator actions	RES/DRPS/RHFB	M	12/88	03/89	03/89	N/A	06/90	11/83
B-55	Improve reliability of target rock safety relief valves	RES/DE/EIB	M	09/88	01/89	N/A	N/A	03/89	11/83
B-61	Allowable ECCS equipment outage periods	RES/DRAA/PRAB	M	06/88	09/88	09/88	N/A	10/88	11/83
I.D.3	Safety system status monitoring	RES/DE/MEB	M	02/88	11/88	10/88	N/A	05/89	11/83
I.D.4	Control room design standard	RES/DRPS/RHFB	M	06/90	12/90	09/90	N/A	09/91	11/83
I.D.5(5)	disturbance analysis systems	RES/DRPS/RHFB	M	Complete closeout memo to EDO				12/87	11/83
II.B.5 (3)	Effect of hydrogen burning and explosions on containment structure	RES/DRAA/AEB	M	12/86C	A	N/A	SA	09/88	11/83

TABLE E.2. (contd)

Issue Number	Title	Office/ Division/ Branch	Priority	Draft Resolution	ACRS Review	CRGR Review	Commission Review	Current Resolution Date	Prioritization Date
II.E.6.1	Test adequacy study	RES/DE/EIB	M	04/89	02/90	02/90	N/A	05/90	11/83
II.F.5	Classification of instrumentation, control, and electrical equipment	RES/	M	Complete closeout memo to EDO				01/88	11/83
66	Steam generator requirements	NRR/DEST/EMTB	NR	04/83C	10/83C	10/83C	09/84C	TBD[a]	11/83
75	Generic implications of ATWS events at the Salem nuclear plant	RES/DRA/ARGIB	NR	09/88	02/89	04/89	N/A	01/90	10/83
83	Control room habitability	RES/DRAA/SAIB	NR	08/89[6]	10/89	12/89	02/90	04/90	06/84
84	CE PORVs	NRR/DEST/SRXB	NR	Deferred[7]				TBD	02/85
86	Long range plan for dealing with stress corrosion cracking in BWR piping	NRR/DEST/EMTB	NR	--	--	08/87	09/87	10/87	10/84
91	Main crankshaft failures in TDI diesel	NRR/DEST/EMTB	NR	Completed				09/87	09/85
102	Human error in events involving wrong unit or wrong train	NRR/DLPQ/LPEB	NR	Integrated into policy statement[3]				TBD	02/85
103	Design for probable maximum precipitation	RES/DE/EIB	NR	01/88	09/88	09/88	N/A	10/88	09/85
124	Auxiliary feedwater system reliability	NRR/DEST/SRXB	NR	Schedule has plant specific milestone dates				02/88	02/86
B-64	Decommissioning of nuclear reactors	RES/DE/MEB	NR	11/86	06/87	06/87	12/87	01/89	11/83
I.D.5(3)	On-line reactor surveillance systems	RES/DE/EMEB	NR	Complete closeout memo to EDO				06/89	11/83

TABLE E.2. (contd)

Issue Number	Title	Office/ Division/ Branch	Priority	Draft Resolution	ACRS Review	CRGR Review	Commission Review	Current Resolution Date	Prioritization Date
II.J.4.1	Revise deficiency report requirements	AEOD/	NR	--	10/87	10/87	11/87	12/87	11/83

(a) Received 11/9/87 from R. C. Emrit NRC/RES/DEA.
(b) Resolution date may be reevaluated depending on outcome of North Anna steam generator tube rupture evaluation that is currently in progress.
A = Annual review; SA = Semi Annual review.
(c) On hold pending resolution of Important to Safety and Safety-Related Issues.
(d) Inspection procedures will be developed based on experience gained during performance of TIs.
(e) The staff is considering dropping two unresolved items and closing this issue with no further action. Based on anticipated transfer of lead responsibility from NRR to RES before 10/01/87. Pending Resolution of Generic Issue A-45.
(f) Issue date for final rule, final resolution schedule will be provided by NRR after transfer.

APPENDIX F

NRC/NRR USER NEED LETTER

APPENDIX F

NRC/NRR USER NEED LETTER

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 0 9 1987

MEMORANDUM FOR: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: USER NEED LETTER - NUCLEAR PLANT AGING RESEARCH PROGRAM

RES is currently formulating a major research effort to study the effects of aging on nuclear power plant structures, systems, and components. We are in full agreement that aging effects are a major concern of this agency and we endorse the general objectives of your program.

My staff has been working closely with yours over the past year and we are currently reviewing the details of your aging program plan. The purpose of this memo is to provide you with some additional guidance regarding the scope of your aging research programs to ensure that their usefulness to the regulatory staff will be maximized.

The usefulness of any research result is in a large part measured by its quantitative safety significance. For aging-related research to be of maximum benefit to NRR, we must know not only the effects of aging on structures, systems, and components, but also the risk significance to public health and safety of the aging process in structures, systems, and components if aging is allowed to proceed uncorrected. Knowledge of potential risk reductions due to various corrective actions, such as maintenance and replacement, would be extremely useful in assisting us to determine proposed regulatory actions. The aging data should also include information which permits extrapolation of the aging process and associated risk significance into a time frame appropriate to the periods expected to be requested for license renewal. We would like your aging program to include these additional elements in it so that research results could be more easily and justifiably implemented into the regulatory process. Our staffs should work closely together regarding the detailed implementation of these suggestions.

I also ask you to consider implementing a program of establishing the risk significance of results in other appropriate research areas.

James H Sniezek / for
Harold R. Denton, Director
Office of Nuclear Reactor Regulation

CONTACT: D. Cleary, EIB, DSRO
49-28547

EEICB, DET, RES PROGRAM FOR

NUCLEAR PLANT AGING RESEARCH/LIFE EXTENSION

Policy and Planning Guidance, Technical Integration, User Needs

J. Vora
EEICB, DET, RES

EEICB, DET, RES PROGRAM FOR
NUCLEAR PLANT AGING RESEARCH/LIFE EXTENSION
Policy and Planning Guidance, Technical Integration, User Needs

A.	Excerpts from the Proposed 1986 Commission's PPG	1986
	A.1. Aging	
	A.2. Plant Life Extension	
	A.3 Mothballing/Reactivation of Nuclear Power Plants	
B.	Technical Integration - "Aging"	
	Memorandum from the Office Directors to EDO	11/13//85
C.	Closeout of Generic Issues B-58 and C-11	
	Memorandum from H. R. Denton to T. P. Speis	07/09/85
D.	RES Assistance on Generic Issue No. 70 - "PORV and Block Valve Reliability"	
	Memorandum from J. P. Knight to G. A. Arlotto	06/27/85
E.	NRR Review of RES FY 1987 Internal Budget Presentation to ACRS and BRG	
	Memorandum from H. R. Denton to R. B. Minogue	05/29/85
F.	Relicensing of "Mothballed" Plants	
	F.1. Memorandum from J. R. Carter to B. M. Morris	04/02/85
	F.2. Memorandum from D. Crutchfield to D. Eisenhut	01/24/85
G.	NRR Input for the Long Range Research Plan (LRRP)	FY 1986-1990
	Memorandum from H. R. Denton to R. B. Minogue	12/06/84

H. DHFS, NRR Maintenance and Surveillance Plan 10/10/84

I. Use of Signature Tracing Techniques to Detect Degradation or Incorrect Adjustment of Safety Related Motor Operated Valves

Memorandum from H. R. Denton to R. B. Minogue 06/21/84

J. Draft Maintenance Program Plan: RES Resources

Memorandum from H. L. Thompson to G. A. Arlotto 05/24/84

K. Review Comments on RES' Program Plan on Nuclear Plant Aging Research

K.1. Memorandum from J. N. Grace, E. L. Jordan to B. M. Morris 06/15/84
K.2. Memorandum from W. T. Russell to L. S. Rubenstein 04/11/84
K.3. Memorandum from K. V. Seyfrit to B. M. Morris 03/01/84

L. Staff Involvement in Shippingport Decommissioning

L.1 Memorandum from Chairman Palladino to EDO 11/29/83
L.2 Memorandum from EDO to Chairman Palladino 12/16/83
L.3 Memorandum from E. L. Jordan to G. A. Arlotto 04/17/84
L.4 Memorandum from D. Eisenhut to R. Mattson, T. Speis, H. Thompson, R. Vollmer 04/19/84

M. Potential Areas of Research in Equipment Qualification

Memorandum from V. S. Noonan to W. F. Anderson 07/15/83

N. NRR's Comments on the Draft RES LRRP for FY 84-88

Memorandum from H. R. Denton to R. B. Minogue 03/25/82

O. ACRS Report to Congress (NUREG-0864) 02/1982

RES Contractors Currently Involved in the NPAR Program Address:

ORNL	-	B0828	A, B, C, D, F, H, I, J, K, M, N, O
PNL	-	B2865	A, B, F, H, J, K, L, N, O
INEL	-	A6389	A, B, E, F, G, H, J, N, O
BNL	-	A3270	A, B, F, H, J, K, N, O

Plant Aging Research/Life Extension

Policy and Planning Guidance, Technical Integration, and Users Needs

A. Excerpts from the Proposed 1986 Commission's PPG

A.1 "The NRC will continue to seek to understand the effects of aging and irradiation of materials and components in reactor containments.

- The staff will conduct research to identify measures which can be taken to correct deficiences attributable to aging and irradiation and to reduce risks inherent to degraded equipment."

A.2 "The Commission intends to begin development of the policies and criteria to define requirements for operating license extensions to help assure that industry's efforts in this area are focused on the primary regulatory concerns.

- In view of industry initiatives to address plant life extensions, the staff should propose policy guidance and develop licensing criteria to define requirements for operating license extensions. The staff should work with industry to ensure that key regulatory issues are identified."

A.3 "The staff should propose policy guidance and develop procedures and requirements for mothballing and for proceeding with such projects."

B. Technical Integration "Aging" (Ref. Memorandum from the Office Directors to EDO dated November 13, 1985)

"(1) Aging - Potential involvement by all major offices. This effort will study such time-related issues as the mechanisms of aging and degradation, methods of examination and testing to determine the condition of components, and interpretation of results of these tests for appropriate action. Work will provide the bases by which the staff can initiate requirements for test and examination of components. These assessments will also provide bases for licensing decisions on whether operating plants continue to meet health and safety requirements and, if needed, will facilitate relicensing to extend plant life beyond that originally anticipated."

C. Closeout of Generic Issues B-58 and C-11 (Ref. Memorandum from H. R. Denton to T. P. Speis, dated 07/09/85)

"The scope of each issue proposes comprehensive, systematic studies with the purpose of identifying significant sources of pump and valve unreliability, and implementing maintenance, redesign or replacement programs. A comprehensive systematic study of this type has not been undertaken thus far in NRR. This type of program is more suitable for Office of Nuclear Regulatory Research (RES) work. Nevertheless, there has been a considerable amount of related pump and valve reliability work performed to date.

Under the Nuclear Plant Aging Research (NPAR) program systematic studies will be performed to (a) identify aging and service wear effects associated with mechanical components that could impair plant safety, and (2) identify techniques that will be effective in determining aging and service wear effects prior to loss of safety function so that proper maintenance and timely repair or replacement can be implemented. Although this is a program intended to encompass many component types it is envisioned to include various safety-related valve types and pump components. RES intends to evaluate LWR operating experience and identify aging trends. One of the specific benefits cited is improved reliability and availability.

From our evaluation of the NPAR program we have concluded that it generally encompasses the scope of the programs inferred by generic issues B-58 and C-11. When this program is completed it is expected that recommendations will be made by RES for maintenance, repair, or replacement, according to component type. At that time these recommendations will be grouped into manageable tasks and considered by NRR staff for making changes to regulatory requirements.

CONCLUSIONS

A substantial amount of work related to pump and valve reliability has already been done to reduce risk by identifying and resolving specific issues. Concerns identified by AEOD related to this subject will continue to be prioritized and worked as appropriate. The aging research program includes systematic studies as called for in B-58 and C-11 and the results will be used by NRR to make appropriate changes in regulatory requirements. On this basis B-58 and C-11 will be closed out and further resources related to these generic issues will only be used to monitor the aging research program."

D. <u>RES Assistance on Generic Issue No. 70- "PORV and Block Valve Reliability"</u> (Res: Memorandum from J. P. Knight to G. A. Arlotto, Dated 06/27/85).

"In recent discussions between DE and RES staff regarding the RES effort to study component aging and service wear, we became aware of a current effort at Oak Ridge National Lab (ORNL) which involves various types of acquisition of operational data which might be of use to NRR in several areas of study. One of the possible uses which we would have for such data is in the resolution of Generic Issue No. 70 (GI-70). As shown in the attached task action plan (TAP), the Mechanical Engineering Branch (MEB) has the lead for completing subtasks 1.1 and 1.2 of GI-70 which involves in part the acquisition of PWR pressurizer PORV and block valve data. Among the items which we believe we could use RES/ORNL assistance in completing these subtasks are in providing the following information: (1) a complete list of the failures and failure rates of PORVs and block valves, (2) component and associated control system failure causes, (3) any other PORV or block valve failure event contributors, and (4) subsequent corrective actions.

We have already discussed this with Mr. J. Vora in RES, NRC Project Manager for the ORNL aging program. Based on Mr. Vora's discussion with ORNL, we understand this work can be started June 1985 under the scope of the current RES contract FIN B-0828. In order for us to be most able to evaluate the information provided, we request that the data be categorized in at least the following three ways: (1) by plant name, (2) by valve vendor, and (3) by failure mechanism. In addition to LER information, Nuclear Plant Reliability Data System (NPRDS) and Foreign Event File (FEF) information should also be used. Because of recent specific comments made by the ACRS Sub-Committee on Valve Reliability, ORNL should pay particular attention to any reported PORV failures resulting from incompatibility with boron in the primary coolant fluid.

Because of the short time period allowed for subtasks 1.1 and 1.2 in the TAP, we request that this work be initiated as soon as possible."

E. NRR Review of RES FY 1987 Internal Budget Presentation to ACRS and BRG (Ref: Memorandum from H. R. Denton to R. B. Minogue, Dated 05/29/85)

"Systems Aging and Service Wear Effects," (A-6389). Furthermore, we would like to see by the end of FY 1986 a summary report which presents the results of this study of whether and where equipment aging is a safety problem."

F. Relicensing of "Mothballed" Plants (Ref: Memorandum from J. R. Conti to B. Morris, Dated 04/02/85; Memorandum from D. Crutchfield to D. Eisenhut, Dated 01/24/85.)

"RES currently has a Nuclear Plant Aging Research Program in progress. RES will be asked to identify any type of components or technical concerns with respect to preservation/ maintenance. This will be used as one aspect of whether a facility or stored equipment in a facility may be reactivated and if so, what testing may be necessary."

"...identify aging concerns that would exist if a reactor facility had the construction delayed or prolonged. Are there any precautions that must be taken if NRC were to ultimately license the facility for operation?"

G. NRR Input for the Long Range Research Plan (LRRP) FY 1986-1990 (Ref: Memorandum from H. R. Denton to R. B. Minogue, Dated 12/06/84)

"We also request that RES initiate a program to develop guidance for license renewal applications. As plants near the end of their operating licenses, we expect to receive applications for operating license renewals or extensions. At present, there is no guidance for making or reviewing such applications. Our preliminary priority ranking is 'medium'."

H. DHFS. NRR Maintenance and Surveillance Program Plan (Ref: Memorandum from H. L. Thompson to R. B. Minogue, Dated 10/10/84)

Problem and Objectives include:

" -- To detect the causes and effects of equipment degradation, and to identify corrective action to minimize equipment failures and unavailability.---"

Technical Issues include:

" o Indicators of maintenance effectiveness
  -- Measures of maintenance effectiveness may include indices of the effects of aging. ---
o The role of preventive maintenance in counteracting aging and service wear effects."

Related Projects include:

---"The nuclear plant aging research of RES/DET will provide information related to the role of maintenance in counteracting the effects of plant aging and make recommendations as appropriate for criteria and standards development."

-3.3.6 Equipment Qualification

---"The equipment qualifications program requires information regarding age-related performance degradation of equipment and components and maintenance and surveillance activities which affect equipment integrity. Results of the nuclear plant aging research will provide implications for the environmental integrity of electrical and mechanical components and equipment qualifications."

I. Use of Signature Tracing Techniques to Detect Degradation or Incorrect Adjustment of Safety Related Motor Operated Valves (Ref: Memorandum from H. R. Denton to R. B. Minogue, Dated 06/21/84)

"Background

Past NRC studies have identified numerous events in which motor operated valves (MOVs) have failed to operated during required plant inservice tests. A significant portion of the events are symptomatic of aging mechanisms and associated with making incorrect valve adjustments. This trend is continuing as is evidenced in current license event reports. These types of valve malfunctions were highlighted in a report recently published by Oak Ridge National Laboratory under an RES funded contract, NUREG/CR-3543 "Survey of Operating Experience from LERs to identify Aging Trends" dated January 1984.

Information Needs

We request that RES provide funding for the expeditious development and implementation of a limited test program utilizing this new equipment. The overall objective of the program should be to learn exactly what the equipment can provide about safety related MOV operational readiness above and beyond the currently used ASME Section XI methods.

We believe that this requested program would be a natural fit with the RES current program "Operating Reactors Inspection, Maintenance and Repair, Nuclear Plant Aging Research" (FIN B-0828). In particular the subtask at ORNL covering detection of defects and degradation monitoring seems appropriate. This request has been discussed with W. Morris and J. Vora of the RES staff who are responsible for the referenced program.

We believe that, in addition to NRR, the results of this program will be of interest to the Office of Inspection and Enforcement and Analysis and Evaluation of Operational Data, and that they should also be kept informed of the progress of the program.

Schedule

Because of the potential for near term improvement in safety-related valve reliability through use of this type of test equipment, we request, funding permitting, that a limited program be undertaken in FY 84 consistent with the medium priority categorization of generic issue II.E.6.1 and scheduled resolution of the issue no later than the end of FY 1986.

Other NRC Office Concurrence

This proposed program has been discussed with the Office of Inspection and Enforcement, Division of Emergency Preparedness and Engineering Response, and they have concurred in this memorandum."

J. Draft Maintenance Program Plan: RES Resources (Ref: Memorandum from H. L. Thompson to G. A. Arlotto, Dated 05/24/84)

"Two RES projects have been identified in the Maintenance Program Plan which appear to be most appropriately conducted as part of your Nuclear Plant Aging Program (NPAP). These are:

3.4 Investigate Relationship between Plant Cycling and Maintenance

3.6 Establish Preventive Maintenance Requirements and Techniques

My staff and yours (J. Vora) have been coordinating these efforts. Please advise as soon as possible whether it is your intent to cover the above Maintenance Program Plan requirements within your existing Plant Aging Porgram or whether additional resources are required."

K. Comments on Nuclear Plant Aging Research Program Plan

K.1 Memorandum from J. N. Grace, E. L. Jordan to B. M. Morris, Dated 06/15/84

K.2 Memorandum from W. T. Russell to L. S. Rubenstein, Dated 04/11/84

K.3 Memorandum from K. V. Seyfeit to B. M. Morris, Dated 03/01/84

Comments in the aforementioned memoranda were resolved - to the satisfaction of the staff from the respective offices (NRR, IE, AEOD).

L. Staff Involvement in Shippingport Decommissioning (Ref: Memorandum from Chairman Palladino to EDO, Dated 11/29/83)

L.1 "I am interested in knowing the extent to which the staff is or is not participating in deciding what data we should be seeking from Shippingport as part of its decommissioning. In particular, I would like to know if each staff office (RES, NRR, I&E, etc.) is giving overt attention to this data gathering opportunity."

L.2 (Ref: Memorandum from EDO to Chairman Palladino, Dated 12/16/83)

"Nuclear Plant Aging Research

Enclosures 1 and 2 represent an exchange of correspondence between Mr. Minogue and Admiral McKee regarding the use of Shippingport for conducting research. Mr. Minogue initiated this correspondence because we have begun a research program to evalute the potential impact on safety associated with aging, including wear, of commercial nuclear plant equipment and structures and believed that Shippingport could be fruitful in supplying important information. A meeting was held on November 30, 1983 at Shippingport to begin the process of selecting equipment which is sufficiently similar to LWR equipment that examination or tests would be beneficial in meeting the objectives of the aging research program. Our staff is staying in close contact with Admiral McKee's staff as we move forward in getting the needed technical information from Shippingport.

Decommissioning Research

RES has also initiated a research program to provide data to allow an assessment of radiation exposure during decommissioning and of implementation of ALARA techniques. Radioactive inventories, exposures, dose rates, contamination levels, ALARA techniques and results, waste shipment and disposal costs and public dose reports will be obtained by observing decommissioning activities at various facilities, including Shippingport. Current NRC activities related to Shippingport include collection of engineering data and preliminary exposure estimated. These will be used for comparison to data obtained during the actual decommissioning which is not expected to begin until late 1984 or early 1985.

Although the other NRC offices are not currently conducting specific data gathering activities related to Shippingport, appropriate technical staff from NRR, IE, and AEOD have participated in the planning of the research activities discussed above and will be participating in the review of the planning and implementation as the research goes forward. This process should provide an adequate mechanism to assure that the staff takes advantage of potential opportunities, in addition to aging and decommissioning studies, for obtaining data from Shippingport."

L.3 (Ref: Memorandum from E. L. Jordan to G. A. Arlotto, Dated 04/17/84)

"We urge that particular attention be directed toward procuring metallurgical samples from the reactor coolant system. These samples would have high value in bench marking metallurgical research now in progress under the auspices of your materials engineering branch. We have discussed this with Mr. Serpan, and he is enthusiastic.

We particularly urge that you obtain sizable samples from the thermal shield and from the reactor vessel in the core belt line region. We recommend that a specimen at least 12-inches in diameter be removed by core drilling through both the thermal shield and the vessel wall. A location should be chosen such that the intersection between a girth and longitudinal weld is included. Sectioning of the specimen would provide an unequalled opportunity to evaluate radiation damage to the vessel all the way through the wall. We are advised that the vessel material is the same as that used in licensed reactor vessels now in service. We recognize that the reactor coolant system operating temperature at Shippingport is slightly lower than the typical commercial plant temperature, but we do not consider that to be a significant impediment. The neutron damage rate in the Shippingport vessel may be somewhat higher but that would only make the results slightly conservative.

We also urge that samples be obtained from at least one loop of the reactor coolant system. These should include the wrought stainless steel pipe and cast stainless steel valve or pump bodies. We understand that some of the pumps may have been replaced since initial construction. The chosen samples should be from original equipment, since our area of interest relates to long-term effects.

Our primary concern at this time is assurance that the necessary samples will be obtained while they are still available. We will be glad to work with your staff in development of an appropriate program and schedule for detailed analytical work."

L.4 (Ref: Memorandumn from D. Eisenhut to R. Mattson, T. Speis, H. Thompson, R. Vollmer, Dated 04/19/84)

"As you may know, the decommissioning of the Shippingport reactor will commence in late 1984 or early 1985. This activity represents an unusual opportunity for gathering useful data regarding light water reactors. The Office of Research (RES) has initiated discussions with the Navy regarding the use of Shippingport for conducting research. RES expects to obtain data useful in both their Nuclear Plant Aging Research Program and Decommissioning Research. In order for NRR to participate and benefit from this data gathering opportunity, we must communicate our needs and desires to RES. Therefore, I request that each division review its technical program (e.g., USI's, generic issues) and identify areas where information and data from plant decommissioning activities may be useful."

M. Potential Areas of Research in Equipment Qualification (Ref: Memorandum from V. S. Noonan to W. F. Anderson, Dated 07/15/83)

"In the Equipment Qualification Branch's continuing review of electrical and mechanical equipment qualification used in operating plants and plants receiving new operating licenses, several technical issues arise. The attachment provides an outline of the areas of concern in connection with seismic and environmental qualification, including pump and valve operability.

B. Surveillance of Age-Related Degradation of Electrical Equipment

BACKGROUND INFORMATION

Equipment installed in operating plants has, in many cases, been qualified to older standards and requirements not requiring the use of accelerated aging prior to exposure to accident conditions. In addition, the artificial aging utilized on newer equipment may not simulate the type and extent of aging degradation experienced under actual plant conditions.

As a result, it is necessary to establish methods for monitoring the condition of equipment located in a harsh environment to determine if significant age-related degradation is occurring.

OBJECTIVE:

Identify practical methods for monitoring and measuring the age degradation of electrical equipment in a harsh environment.

SCOPE

1. Identify equipment expected to experience a significant age-related degradation and which is located in a harsh environment.

2. By use of experience data, previous testing and analysis, identify the most promising methods for measuring degradation, including an indication of precursors to failure modes.

3. Conduct testing to improving the understanding of the phenomenological process to provide visibility to degradation.

4. Recommend guidelines and criteria for use by the licensing staff to assess if the applicant's program is adequate to assure function is not compromised during and after an accident."

N. NRR Comments on the Draft Long Range Research Plan for FY 84-88
(Ref: Memorandum from H. R. Denton to R. B. Minogue, Dated 03/25/82)

"We have reviewed the problem areas of concern to NRR that could benefit from research early in the FY 84-88 time frame, and have identified the following high priority areas:

Research into aging of plant structures, systems, and components, including material degradation, valve behavior, flaw detection, maintenance, and inservice inspection."

O. Report to Congress (NUREG-0864 Safety Research Program for FY 1983
3.4 Aging

As nuclear plant operating life advances, some degradation in its equipment and systems must be expected. Pressure vessel embrittlement has been recognized for many years to increase with accumulated fast neutron fluence requiring adjustment of the operating temperature and pressure limits to assure rupture resistance. The frequency of steam generator tube leakage also increases with operating life. Organic coverings of electrical and control cabling, elastomer seals on equipment closures, and potting compounds for electrical connections are all known to deteriorate under extended exposure to humidity and temperature conditions and low level gamma and beta irradiation. Cyclic operating conditions affect the performance reliability of valves, motor operators, ventilation machinery, and comparable equipment. Emergency diesel power equipment and DC power sources also are subject to degradation with operating use. The safety of nuclear power plants depends on making certain that such degradation with age is recognized and accommodated before it can cause significant reduction in safety.

There is inadequate knowledge regarding the effectiveness of the surveillance programs related to aging and the effects of operating transients on equipment that has deteriorated with age. As highlighted by the recent concern for thermal shock on some older pressure vessels, it appears important to apply some research effort to improve understanding of the safety significance of loss of capability through such aging. The NRC should initiate a comprehensive, systematic investigation of safety-related effects of aging for LWRs."

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The "Plan for Integration of Aging and Life Extension," developed by Technical Integration Review Group for Aging and Life Extension (TIRGALEX) in May 1987, identified the safety-related nuclear power plant structures and components (S/C) that should be prioritized for further evaluation by the NRC's Nuclear Plant Aging Research Program (NPAR).

This report documents the results of an expert panel workshop established to perform the S/C prioritization activity. Prioritization was primarily based upon criteria derived from a specially-developed risk-based methodology. This methodology incorporates the effect upon plant risk of both component aging and the effectiveness of current industry aging management practices in mitigating that aging.

An additional set of criteria used to categorize the S/C is the importance of aging research on S/Cs to the resolution of generic safety issues (GSI) and/or to identify NRC/NRR user needs. The resultant S/C categorization was to provide additional information to decision makers, but was not used to calculate final S/C ranks.

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