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Technical Report

AGING AND LIFE EXTENSION ASSESSMENT PROGRAM (ALEAP)

SYSTEMS LEVEL PLAN

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FOREWORD

This plan for aging assessment at the systems level has been prepared in a generic fashion such that it can be used by any of the NRC aging program participants, if so desired. It will be used by Brookhaven National Laboratory for their FY 1987 aging assessment of the following two systems: 1) Component Cooling Water, and 2) Residual Heat Removal (RHR) low pressure emergency core cooling.

The follow-on work, referred to as Phase II, is not authorized by the NRC at this time. However, it is a logical extension of the Phase I tasks, and is in accordance with the goals and objectives of the NPAR program plan - NUREG-1144.

To avoid duplication of effort, and to maintain the high quality output of technical work, it is essential that close coordination be maintained among all of the NPAR national laboratory participants. This is particularly important in the areas of failure data analysis and PRA modeling.

Finally, a discussion of life extension issues has been included. This was done because an understanding of the aging process is essential in any plant life extension analysis, and conversely, an awareness of the ultimate utilization of NPAR program outputs (e.g., for life extension) will be helpful in shaping the tasks that are presently at hand.

ACKNOWLEDGEMENT

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EXECUTIVE SUMMARY

This Systems Level Program Plan for ALEAP presents and explains the BNL structured approach to assessing the effects of the aging of nuclear power plant (NPP) components and systems on the safe operation of NPPs and the extension of plant operation beyond the originally planned plant life. Note that this plan is prepared in a generic fashion and could be used by anyone for a systems assessment.

The plan discusses the criteria for the prioritization of plant, system, and component selection for analysis to determine the effects of aging. The use of Failure Modes and Effects Analysis in conjunction with the results of natural and accelerated aging tests are discussed as means to understanding and predicting the phenomena. The effects of aging on the failure rates of components will be determined principally from plant data with physical and phenomenological models used for interpolation of areas not included in the data base. These results will be integrated with a plant risk model to be used in addressing the question regarding "how old is old enough."

The NRC Nuclear Plant Aging Research (NPAR) program has completed several component level aging assessments which include the identification of dominant component failure modes based on plant operating experience. The studies provide recommendations for monitoring, as well as mitigating, age-related component degradations.

Utilizing results from the component level studies, and work performed by other NRC contractors for systems data assessment and systems level risk analysis, this program will evaluate the impact of component failures on plant system performance. The study will perform in-depth systems level failure data reviews, reviews of current industry practices for system maintenance, testing and operation, and probabilistic risk assessment (PRA) techniques to understand and to predict the impact of aging on system availability. Recommendations for improving the system performance by means of degradation monitoring and timely preventive and corrective maintenance will be addressed. This project will integrate its products with the BNL programs for Operational Safety Reliability Research and Performance Indicators.

The first phase of this research effort will concentrate on understanding various system designs from plant safety analysis reports, evaluating failure data from plant operating experience data bases, applying PRA analyses, assessing industry wide surveillance and maintenance practices, and identifying system functional indicators which are used to monitor the rate of system degradation resulting from aging and service wear. The program will separate failures on demand from time-dependent failures. It will categorize separately, age-related failures from random and design type failures. It will produce results useful for resolution of pertinent unresolved safety issues and for review and inspection of operating NPPs. The second phase, if authorized and performed, will provide recommendations for improving the system performance through enhanced maintenance practices and reliability monitoring which will be focussed on the most risk sensitive areas of a system. Recommendations will be made for improvements in pertinent Regulatory Guides, Industry Standards, etc. This program plan delineates the goals and major tasks to be completed in each phase. The current version of the program plan is considered to be a draft and will be revised and updated as the first few systems are completed using this methodology. This will produce a final proven methodology, which can be used for all remaining systems.

1. INTRODUCTION

1.1 Statement of the Problem

As Nuclear Power Plants (NPP) age, the likelihood of common cause failure due to age-related degradation increases. As a result of the aging of components in a nuclear power plant steps must be taken to: a) assure that the level of safety on which a plant was originally licensed has not degraded below an acceptable level, b) identify modifications, procedures, and maintenance that will arrest or suppress aging effects and restore the reliability to that on which the original license was based, c) recommend incipient failure detection methods to provide early warning of impending failure, and d) determine those factors that must be implemented or evaluated for consideration of NPP life-extension requests.

The Technical Integration Review Group for Aging and Life Extension (TIRGALEX) defines an effective NRC program presenting a structured approach to integrating aging research and regulation. This plan indicates the close-coupling between aging and life extension. The ALEAP plan will be updated as necessary to agree with the TIRGALEX goals.

1.2 Definitions of Aging

To clarify the ALEAP scope and approach, a definition of aging is needed. NPAR (NUREG-1144)[36] defines aging as the "cumulative degradation occurring within a component, structure or system which, if unchecked, may result in loss of function and impairment of safety." Factors causing aging/degradation may include:

- natural internal chemical or physical processes,
- external stresses and environment,
- service wear (cycling, vibrations),
- testing, and
- improper installation, application, and maintenance.

Note the emphasis is on actual physical changes in the properties and performance of the plant and equipment.

Backfits and design changes may introduce new aging mechanisms or special aging concerns, or unforeseen common cause and system interaction problems. An example might be the installation of added electronic equipment in a controlled air conditioned environment resulting in an additional heat load that compromises the temperature control. Similarly, replacement of a pump with one of larger capacity in an auxiliary system may lead to degradation of interrelated piping components due to water hammer or erosion[3].

Life extension is defined as a set of actions and activities aimed at increasing the useful lifetime of a plant or of specific equipment beyond the time originally envisioned. This would be partially based on developed Aging Technology.

The pragmatic concept of aging used in ALEAP is that aging is a process that causes the failure rate of equipment to increase with time. This distinguishes between the "new" performance of equipment when the failures are random (Poisson process uncorrelated with time) and the non-Poisson aging process where failures increase with time. Thus this investigation involves separating age-related phenomena from "like new" failure phenomena, which is taken as the base case. This requires determining the aging fraction of the failure rate as well as the rate of increase. Such information is sought on all levels: plant, safety function, system and component but most of the information is currently available at only the component level as a result of NPAR and data base activities. However, given the age-dependence of components, the dependence of the higher structures formed of the components may be found through PRA system models.

Further analysis of failures at the component level does not necessarily get one to the root cause of failure because components may fail from various causes as they are operated upon by operating and environmental stresses. An understanding of these root cause aging failure mechanisms provides the key to modeling the rate of increase of the aging phenomena as well as the necessary information for mitigating the aging effect. NPAR studies have identified stresses for each component reviewed. Table 1-1 taken from Drago[4], also correlates component types with the stresses that singly or collectively may lead to failure.

These stresses suggest a connection between the environmental qualification of equipment (EQ) as defined in IEEE-Std-323 for equipment in general and other guidance provided in IEEE-Std-382 for valves, IEEE-Std-334 for motors, and IEEE-Std-317 for electrical penetrations as well as Regulatory Guides 1.40, 1.63, 1.73, and 1.131. The connection is that environmental stresses provide many of the degrading mechanisms that cause aging. Process upsets such as addressed in EQ may result in rapid aging for a short time period while normal aging results from operation in a nominal environment for a much longer time but cumulative effects of the normal environment may be much greater than the upset effects.

1.3 Scope of Aging and the Need for Life Extension

ORNL[5] identifies about 17% of the abnormal operating events reported to the NRC as having age-related causes. About 8% of these events resulted from instrument "drift" of the setpoint or calibration outside of the technical specifications. The other nine percent were attributed to aging causes such as wear, corrosion, oxidation, crud deposition and fatigue. Of the components that failed (9% of above events) due to aging, 20% were valves, 14% pumps, 5% diesel generators, 3% steam generator tubes, 3% heat exchangers and less than 1% each for about 120 other components. It should be noted that these data are not normalized to the number of components at risk, but represent the population distribution that would be observed in examining plant data.

Table 1-1. Stresses on Components in a Nuclear Power Plant[4]

Component Types	Stresses*
Accumulators, tanks	T,M,C,H
Air dryers	T,M
Annunicator modules	E,H
Batteries	E,C,H
Blowers, fans, compressors	M,E,V
Battery chargers	E,M
Circuit breakers, motor starters, fuses	E,H
Control rods	M,R,V
Control rod drive mechanisms	M,R,W,V
Demineralizers	C
Electric connectors (cable, bus, wires)	E,R,C
Internal combustion engines	M,T
Filters, strainers, screens	W
Fuel elements	R,V,T,M,C
Generators, inverters	E,M,V
Electric heaters	T,E
Lifting devices (cranes,hoists,jacks)	M,T,R
Heat exchangers (coolers, heaters, steam generators, evaporators)	T,M,C,R,H
Instruments, controls, sensors	E,H,M
Mechanical function units (gear boxes)	M
Motors (electric, hydraulic, pneumatic)	E,M
Penetrations, air locks, hatches	M
Pipes, fittings	C,M,V
Pumps	M,V
Recombiners	C
Relays	E,H
Shocks suppressors and supports	M
Switchgear, load control centers, motor control centers, panel boards	E
Transformers	E
Valves	m,c
Pressure vessels (reactor vessels, pressurizers)	R,T,M,C

*Stress codes - thermal (T), mechanical (M), radiation (R), humidity (H), wear (W), electrical (E), vibration (V), chemical reactions (C). Additional stresses beyond these are also possible.

The data base used in the ORNL study was limited but more extensive information is available from individual plant and industry-wide data collections. Continuing support of industry-wide data collections providing the information for determining the effects of aging must be encouraged. However, in data analysis there is the tradeoff between specificity and statistical accuracy. To aid in overcoming these data deficiencies, it may be possible to use physical and phenomenological models for extrapolating data.

To gain a perspective regarding potential requests for life extension, the paper by Marin[6] indicates that the first license expiration will occur in 1997 and between 2005 and 2010 about 50 GWe or about 75% of the 1983 generating capacity will expire. Figure 1-1 presents a plot from this paper showing the cumulative expirations. Some safety systems and support systems may not be issues in life extension as they may be reparied, replaced, etc.

A overall document on aging mechanisms and aging rates is Carfagno and Gibson[7]. This document provides a valuable review and compilation of physical aging theories and scaling parameters. It also presents phenomenological scaling rules such as those in MIL-HDBK-217. More specialized work addressing the aging of metals is presented by Simonen[8] on embrittlement, by Vignes[9] and by Sanoh[10]. Hinton[11] provides work on the life extension of piping systems and Moelling and Gallucci[12] on probabilistic analysis of stress corrosion cracking in BWRs. Similar aging effects in pipe cracking are provided by Gordon and Gordon[13] on the basic properties of types 304 and 316 stainless steel by Horak[14], and on main feedwater spray heads by Spond[15]. Additional work on the irradiation aging of pressure vessels is presented in Odette[16] aging of piping by Banford[17], and fatigue aging by Server[18]. More comprehensive work on the aging and service wear effects on hydraulic and mechanical snubbers is reported by Bush[19]. Wear measurements of nuclear power plant components is reported by Duframe[20].

Vause[21] reports on the operating experience relating to the aging of diesel generators. Similar work is reported by Dingee and Johnson[22], by Higgins and Subudhi[23], and by Vesely and DeMoss[24]. The aging effects on electric motors has been studied by Subudhi[25], and recommended maintenance practices for life extension by Subudhi[26]. Taylor, et al[27] presents the results of simulated seismic testing of naturally aged small electric motors. Subudhi[28] also presents compiled operating experience and an aging-seismic assessment of electric motors. The results of a correlation study on Class 1E equipment is reported by Sugarman[29]. Bonzon and Hente[30] present their work on seismic fragility tests of Class 1E battery cells. The aging and service wear of check valves is described by Greenstreet[31]. Subudhi and Taylor[32] report their work on reactor coolant pump seals.

On the electrical side, Toman[33] and his associates, present their work on the interactive effects of relay and circuit breaker aging. Gunther[34] indicates the results of work on operating experience and aging-seismic of battery chargers and inverters. Stuetzer[35] is a status report on electrical cable failures due to aging and service wear.

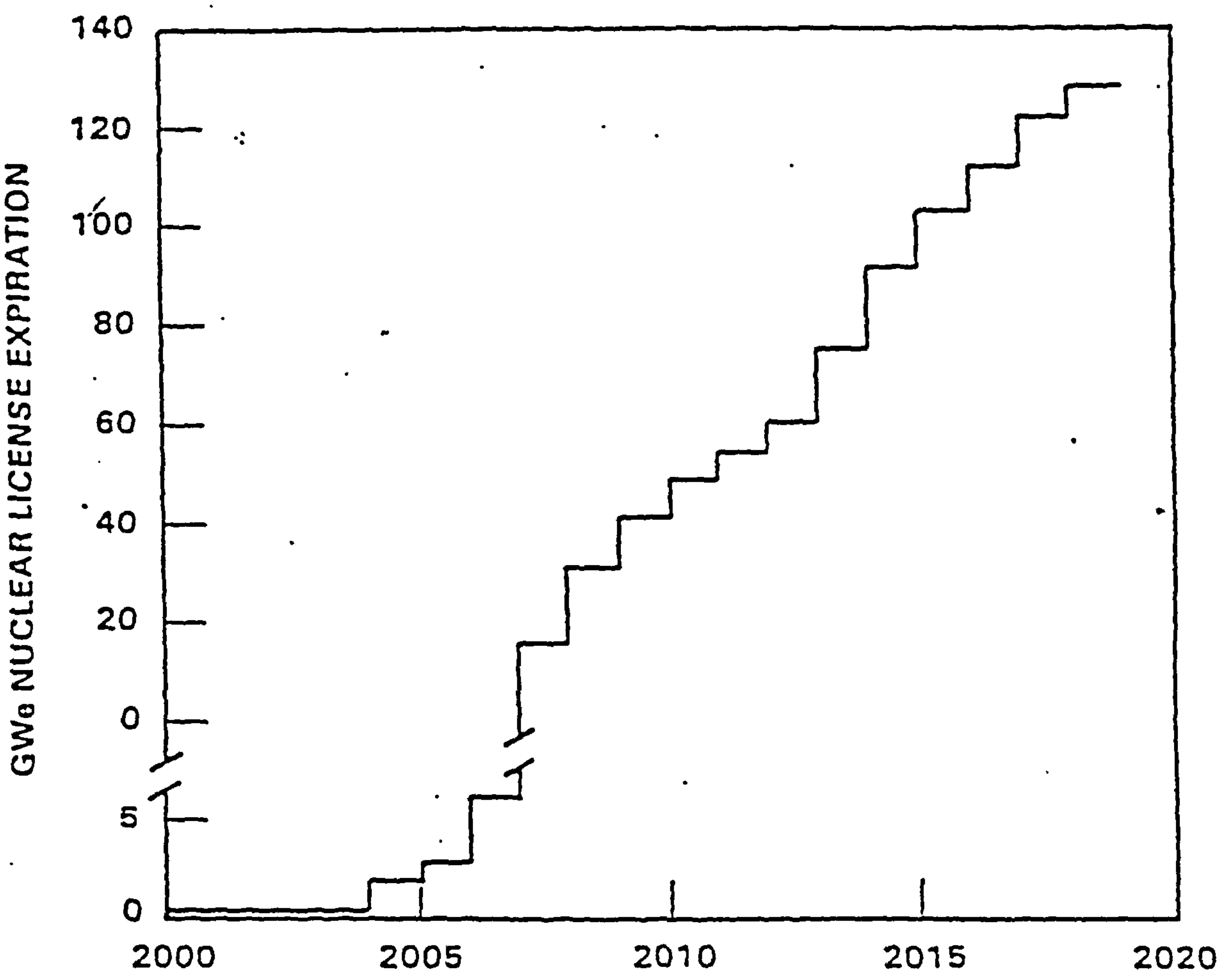

Figure 1-1 Nuclear Power Date Plant License Expirations

Much of the previously described work was performed under the NPAR (USNRC, 1985[36]) program. This work will be used in ALEAP for identifying failure modes and mechanisms of component failures. This will be related to the systems they are contained in, in order to develop system level insights.

Fullwood[37] provides a review of incipient failure detection methods. This report is aimed at reliability improvement but these methods are equally useful for the detection of aging. Sliter and Cady[38] describes EPRI work to this end, as do the papers by Plumstead and Cady[39]; Engh and Figlhubur[40]; and by Weber[41].

A general review of aged power plant facilities is presented in Rose[42]. In Leverenz[43] importance measures are presented for aged components. A report closely related to ALEAP is Vesely[24] in which the rate of aging of selected components is determined from plant failure experience.

2. GOALS AND OBJECTIVES

2.1 Goals

Commensurate with the NRC Nuclear Plant Aging Research (NPAR) program plan, the ALEAP system level plan has the following top level goals:

I. To assess aging impact on system, plant safety, and risk,

II. To develop recommendations to mitigate aging effects to assure public health and safety consistent with plant optimum performance, and

III. To provide initial technical basis for evaluating plant life extension and support regulatory actions.

2.2 Objectives

To achieve the above goals, the scope of the system study is divided into two distinct phases. The objectives of each phase are given as:

Phase I Objectives

1. To identify and characterize the aging impacts on system performance and hence plant safety and risk.
2. To produce interim aging and system related outputs in a form useful for NRR, I&E, and the NRC regions. (See paragraph 3.2.)
3. To address in an interim fashion generic issues related to the systems under study.
4. To assess current inspection, surveillance, and monitoring programs for systems.
5. To assess current maintenance, storage, and mothballing programs for systems and components.

Phase II Objectives

1. To support regulatory actions, as necessary for NRR.
2. To aid in plant life extension decisions.
3. To produce aging and system related outputs in a form useful for NRR, I&E, and the NRC regions.
4. To develop appropriate recommendations to improve inspection surveillance, and monitoring programs.

5. To develop recommendations to improve maintenance, storage, and mothballing programs.

6. To finalize applications to generic issues related to systems under study.

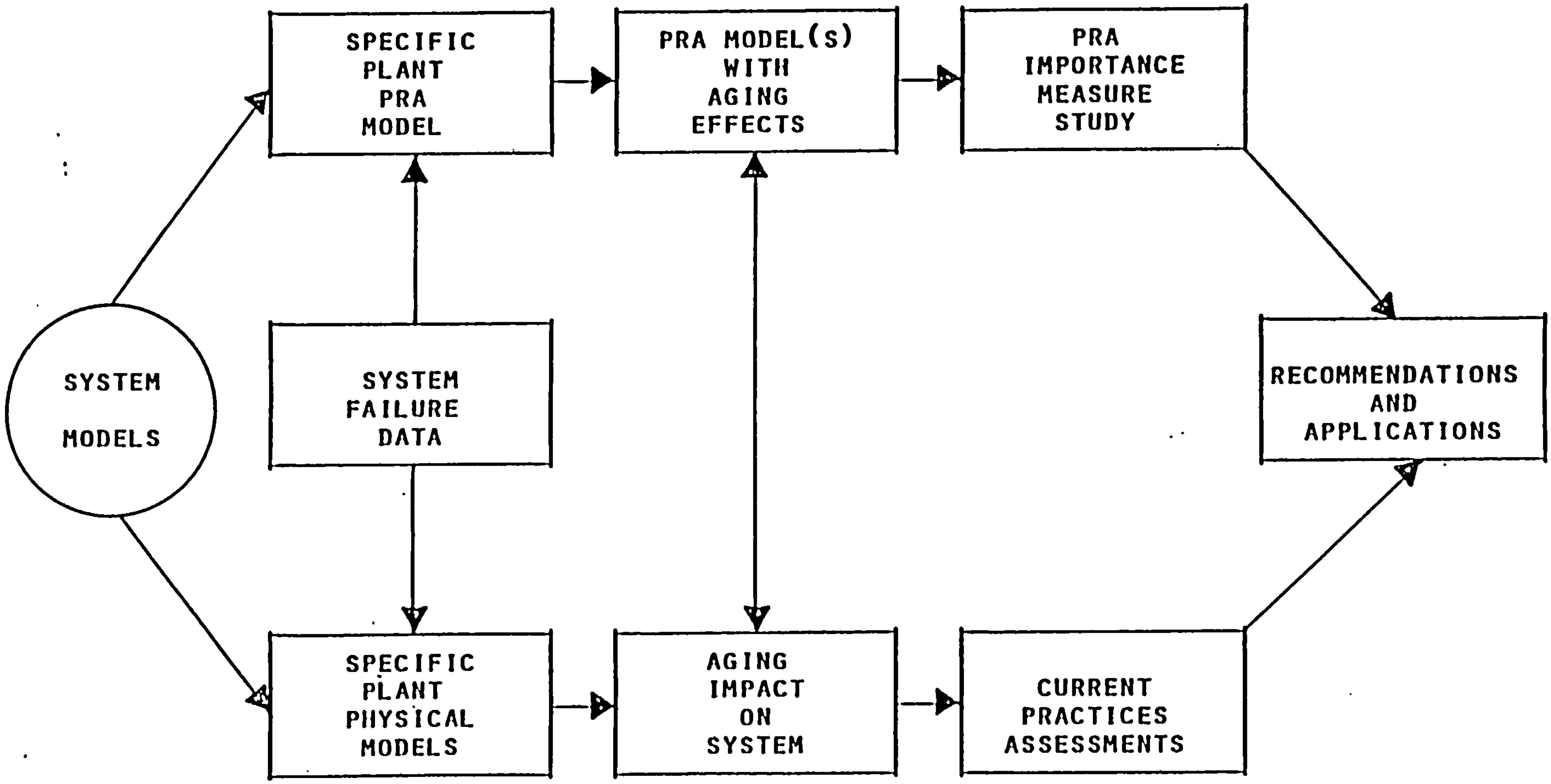

Figure 3-1 Overall Strategy

3. STRATEGY

The ALEAP Systems Level Plan is structured in a two phase approach, with the first phase characterizing the aging effects on system performance and the second phase developing mitigating actions for arresting these adverse effects, prior to system degradation. For the selected system, the phase I study consists of three major tasks: 1) a review of the various systems designs in US nuclear power plants to determine the general applicability of aging analyses performed on a specific plant to other plants, 2) modification and adaptation of existing PRAs such as the NUREG-1150, IREP, RSSMAP, ASEP, and industry-performed to include aging effects, and 3) perform a detailed review of the several data bases and the NPAR results to obtain the primary failure modes, causes, and mechanisms. To perform the multiple computations that will be necessary in the study of aging effects and to calculate the component importances, it is advantageous to have the complete PRA implemented on a computer. An example is the BNL NSPKTR code which models Indian Point and is one of the reasons for choosing this plant for the demonstration. Another example is the PRISIM code that performs calculations on the IREP model of ANO1 which has been used for aging studies of the Auxiliary Feedwater System by INEL. As indicated in Figure 3-1, the failure data analysis results will be fed into both probabilistic and actual system models to characterize the aging effects in the system performance. Once the system behavior is determined, the Phase II study will review and analyze current industry practices for monitoring system performance and mitigating aging effects. Recommendations will be developed for appropriate system monitoring and mitigation techniques in order to improve the system reliability and alleviate aging. A generic schedule for a typical system based on Figure 3-1 is included as Figure 3-2.

In order to achieve the defined goals and objectives the above strategy will be implemented for two systems (CCW and RHR) in FY 87.

Figure 3-3 illustrates all tasks to be carried out in each phase of the program plan. The tasks to be completed and which are included in Figure 3-3, are listed below for further clarification:

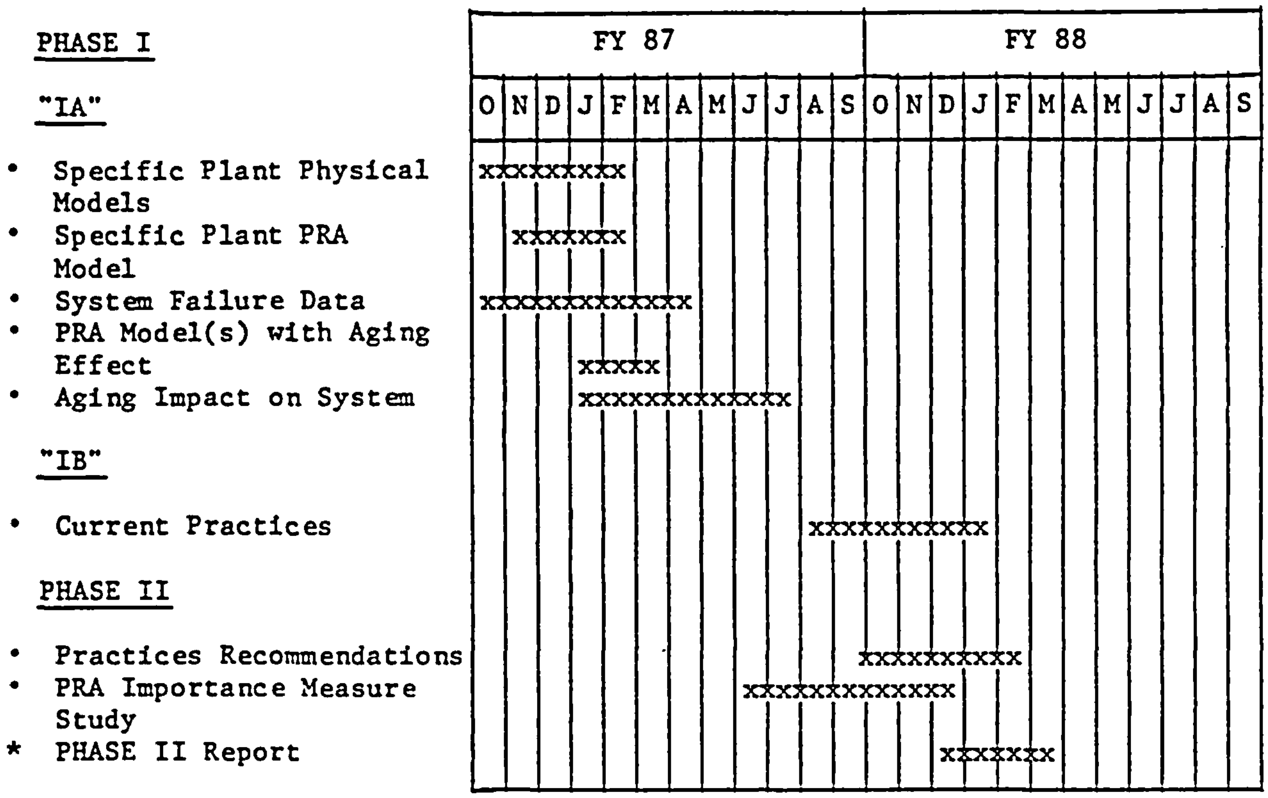

Figure 3-2 Typical Schedule for a NPP System Evaluation

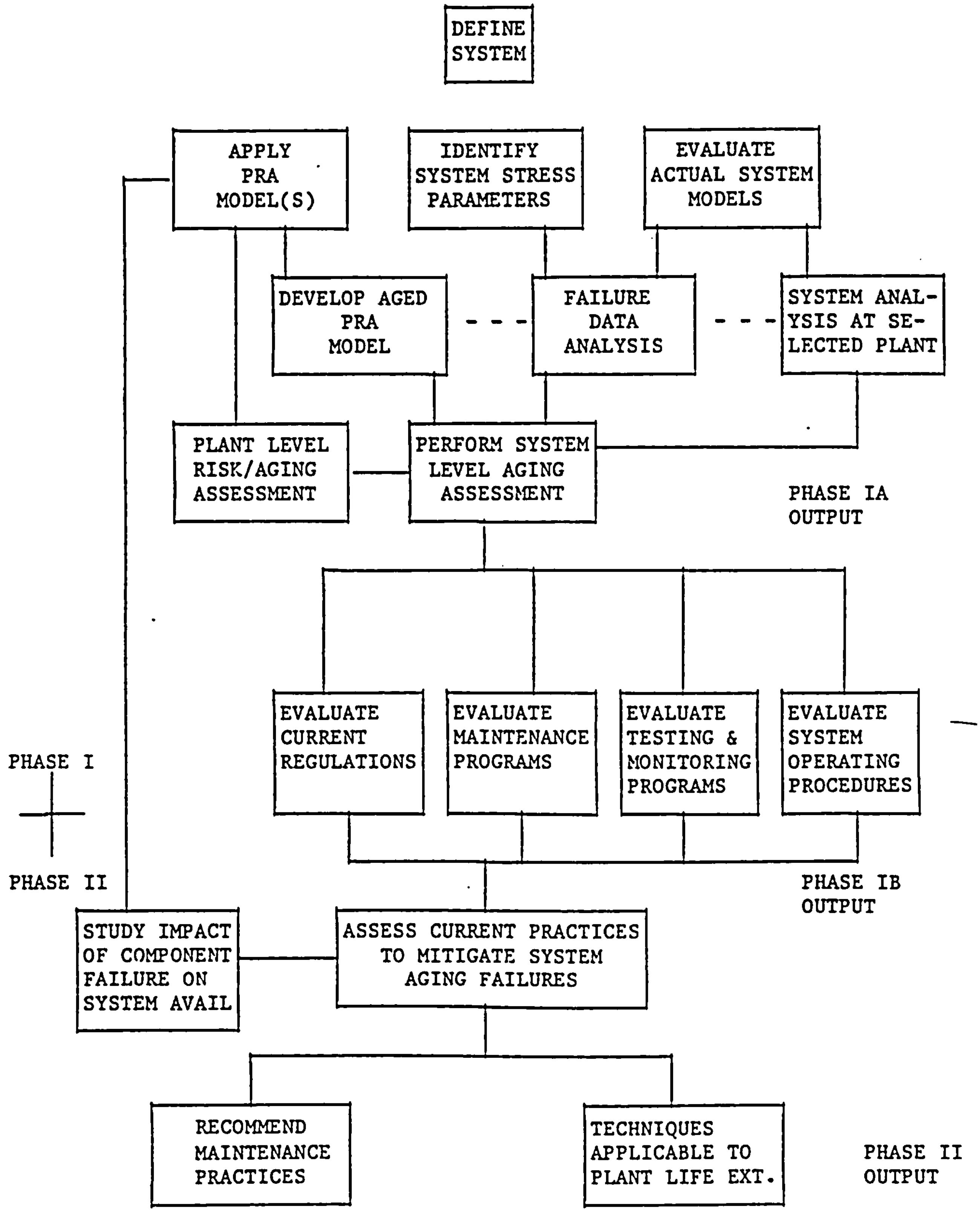

Figure 3-3 Detailed Task Structure of System Level Plan

Phase I Tasks

Phase IA

1) System Definition
2) PRA System Model Application
3) Operating and Environmental Parameters Identification
4) System Analysis
5) Aged PRA Model Development
6) Operating Experience (Failure) Data Analysis
7) System Analysis at Selected Plant
8) System Level Aging Assessment
9) Plant Level Risk Aging Assessment

Phase IB

1) Current Regulations and Guidance Assessment
2) Current Maintenance Practices Assessment
3) Assessment of Current Techniques for Testing and Monitoring
4) System Operating Procedure Evaluation
5) Integration of Tasks 1) Through 4)

Phase II Tasks

1) Plant Risk Assessment
2) Recommended Practices
3) Utilization for Plant Life Extension

4. MAJOR TASKS

This section provides a detailed description of each task. The discussion contains the objectives of the task, input information needed in order to perform the task, and the product of the task. The interrelation among various tasks is illustrated in Figures 3-1 and 3-2. These tasks will be performed for each NPP system. The first two systems to be analyzed by BNL will be the PWR Component Cooling Water System and the BWR Residual Heat Removal System. Upon completion of two systems studies, the draft program plan will be revised and updated to incorporate lessons learned.

4.1 Phase I Tasks

4.1.1 Phase IA Tasks

Task (1): System Definition

The boundaries and the interfaces of the selected system with other plant systems must be clearly defined in order to perform a complete aging assessment. All components and sub-systems that will be studied within each system will be identified. Schematic diagrams showing the interfaces with other mechanical, electrical, instrumentation, and control systems will be developed. Structures supporting components within the system will also be discussed. Assessments in other tasks relating to system performance will be limited to those components within the system boundary. Of particular interest are the way components interact within the system. Design implications at the system interfaces will be discussed, both at the load side and the input or support system side. The interaction between systems at these interfaces will be studied in the subsequent tasks.

A representative plant will be selected for each in-depth systems analysis which will include: review of all design information, review of operating and maintenance procedures and practices, use of plant specific PRA model, and review of all actual plant failure data. The plant selected should be at least 10 years old, have a full scope PRA, be reasonably close to BNL, and be operated by a utility willing to cooperate with the study and share information.

Task (2): PRA System Model Development

New PRAs will not be developed in ALEAP but existing ones such as the recently completed models reported in NUREG-1150, utility performed or sponsored PRAs, WASH-1400, RSSMAP, IREP and/or the ASEP models. Generally speaking, all of these require adaptation to the aging analysis to permit the investigation of the aging-caused change in the failure rates and to determine the importance of the various components. Some PRAs have included importance measure calculations but some of these importance measures are not suitable for aging investigations because the importance measure is not affected by the age of the component being investigated. Extensive work has been done in this area by Vesely for NPAR and will be utilized as a starting point for further work in this program. Calculating the effects of aging and importance calculations, requires repeated calculations of the PRA which suggests a complete computer implementation of the PRA. With some exceptions, PRAs have consisted of multiple separate calculations that are pieced together to obtain a final result for the base case. The exceptions are the BNL PC-code NSPKTR that implements the Indian Point PRA and the PRISIM code, operable on a special PC that implements the IREP model of ANO1. The availability of the NSPKTR code in conjunction with past experience with Indian Point systems, are the reasons for suggesting the use of the CCW at Indian Point for a demonstration. The selection of a plant for the RHR demonstration analysis has not been done but a plant having a PRA that facilitates repeated calculations will be a consideration.

Because of plant and system complexity, it will not be possible to apply aging analysis to all components. A group of components will be selected on the basis of their importance to safety and on the importance of aging effects on their reliability. There are many measures of importance but it seems that the measure "Inspection Importance" previously used in prioritizing NRC inspections is most suitable because it is weighted both by the effect on plant risk if a given component fails and the probability of the component failing. This carries the tacit assumption that components with a high failure rate also have a high aging rate. This is used as a first order selection criterion on which to iterate as the results of aging analysis is incorporated. If this assumption is not completely correct, it will be modified in the reanalysis. In summary, the PRA is used to initially calculate the Inspection Importance of the systems and components. This leads to the systems and their components on which to focus the aging analysis. In the case of Indian Point, the NSPKTR code has preliminarily determined the importance of the systems and shown that the CCW is one of the most importance systems. Subsequent work shown in Appendix B calculates the non-aged importance of the components making up this system. This is followed by an Aging Failure Modes and Effects Analysis (AFMEA - See Table B-5) which is used in conjunction with NPAR work to identify the aging mechanisms which when used with field data or aging phenomenological models provides the age dependence of the component failure rates. These results are fed back into the computerized PRA model for a second iteration so that now the importance measures as well as the reliability and risk assessment contain the effects of aging.

Task (3): Operating and Environmental Parameters Identification

For the aging assessment it is imperative that system physical parameters such as temperature, pressure, humidity, radiation, mechanical and electrical stresses, that affect system performance are identified for both the component and the system level. Typical mechanisms which cause component and system degradations include fatigue cycles (thermal, mechanical, or electrical) wear, corrosion, embrittlement, diffusion, chemical reactions, cracking or fracture, and other overstress mechanisms. This task will analyze all operating modes of the system under normal, abnormal, accident, and post-accident conditions including plant mechanical and electrical transients which contribute significantly to the aging process.

In addition to the operating parameters, environmental conditions are equally important for component degradation. Since the system level analysis will include components both inside and outside the containment, and during normal and accident situations, parts of the system will experience different environmental conditions. Sometimes atmospheric conditions due to plant location require additional analysis for assessing the system failures.

The source of this information will be taken from the plant FSAR, PRA studies, and other plant specific design drawings. Each piece of equipment within the system boundary will be analyzed for the internal and external conditions to assess the aging deterioration of its subcomponents. The output of this task will be the aging characteristic of each component when subjected to its particular operating conditions. NPAR studies on systems and components will be used for the final assessment.

Task (4): System Analysis

This task will review, in detail, the design and specifications of the system under study. Included will be system function, components, and instrumentation. For a support system (such as Component Cooling Water) where there are significant variations between plants, the various system designs must be catalogued and understood. See Appendix C for a sample system survey. The effect of system failure on supplied loads and on overall plant safety/risk must also be taken into consideration. Thereby when the failure data is analyzed both the system design variations and consequences of component failure and degraded system operation will be appropriately treated. Relationship of the system and system problems to pertinent unresolved safety issues must be defined at this stage for later resolution.

The required performance of the system in the various postulated accidents and transients must be clearly understood. Potential failure modes in these scenarios must be considered carefully since they may not be adequately represented in the failure data bases.

Task (5): Aged PRA Model Development

Upon completion of the AFMEA which includes the NPAR work, environmental, service and other causes of aging, it is necessary to quantify the aging effect. This will be done by: a) determining the aging effects that may be observed in field data, or b) physical and/or phenomenological models of the aging process. This work will be closely coordinated with work at other laboratories to avoid duplication. Especially valuable should be the root cause analyses and the investigations of aging dependence that may be obtained from generic data at INEL. These data as well as data developed in this work will be used to determine the contributions of each of the aging processes.

These aging processes will be incorporated with the non-aging processes to provide aging failure rates. These will then be used in the PRA models to determine if the prioritization changes when the aging effects are included in the model. If so, new importances will be calculated that will result in different prioritization. Then the aging FMEA, will be repeated and new age-dependent failure rates calculated. When the age-dependent failure rates are determined to be valid, the plant risk and system availabilities will be recalculated to exhibit the aging effects. If the plant risk is increased above the safety goals or other criteria, the PRA will be re-examined to determine steps for arresting the aging effect.

Task (6): Operating Experience (Failure) Data Analysis

System and component level failure data bases from LER, NPRDS, IPRDS, NPE, completed studies by INEL or ORNL, and other sources will be obtained and evaluated for identifying all failure modes, causes, and mechanisms of components contributing to system failure. This labor-intensive effort will be used to identify critical components, dominant failure sequences, failure mitigation processes, and other relevant information available in the data bases.

Results from this analysis will be fed into the PRA models, as well as to the specific plant models to evaluate the overall system performance and its effect on plant risk. Interim results useful for NRR and I&E systems analysis and inspection will be produced at this stage.

Task (7): System Analysis at Selected Plant

This task will consist of a very detailed analysis of the system under study at a selected, representative nuclear power plant (NPP), plus shorter reviews at one or more than NPPs. BNL will develop close working relationships with one or two local NPP utilities in order to exchange information, and to further the research in the areas under study. Specifically, this analysis will consist of a review of system design, failure data, maintenance records, system testing and operation, and procedures. This will allow BNL to understand how the systems are actually operated and maintained and will correct for deficiencies in the various data bases used in the failure analysis. This task will also allow BNL to learn developing problems in systems under study, and what actions utilities may be taking to ensure proper system operation. Input from actual current plant experience is vital to any study of this nature and this task will provide the needed information. Some information gathered, particularly on maintenance, surveillance, and condition monitoring will be directly used in Phase II of the project.

Task (8): System Level Aging Assessment

Both PRA and plant design models in conjunction with the failure data evaluation will be utilized to assess overall system level aging deterioration. This should provide a prioritized list of components within the system, which may require indepth engineering analysis and better monitoring programs. Critical components based on system unavailability, will be identified. The impact of interface system or components on the subject system's performance will be discussed. This assessment will include all associated categories under mechanical, structural, electrical, instrumentation and control components which contribute to the system failure. Functional indicators for monitoring system degradation will be established.

Task (9): Plant Level Risk Assessment

Using the PC-based plant PRA model, the impact of system unavailability is easily generalized to overall plant risk for assessment using both core melt frequency and offsite consequences as measures of risk. Coordination will be maintained with Vesely and INEL. The component prioritization based on plant risk or core melt frequency, will be obtained considering age of the components as a factor. As the plant ages, the component prioritization may change because of the age-related degradation of the equipment. Plant level performance indicators will be developed to monitor the system health as the plant ages. Consideration must be given to generalizing the conclusions developed here (with the PRA model) to other NPPs, that do not have PRAs.

4.1.2 Phase IB Tasks

Task (1): Current Regulations and Guidance Assessment

This task will evaluate the current status of existing regulatory requirements and industry guidance (including IEEE and ASME Standards) related to the system under study. Included are standards, guides, and NRC regulatory and inspection procedures relating to the subject system. Other NRC related activities such as the maintenance and surveillance programs including plant tech spec requirements, inspection and enforcement activities, plant audit reports, and I&E bulletins and information notices will be evaluated for improvements to mitigate system failures identified in the Phase I study.

In addition to NRC activities, different industry and engineering society activities and standards such as ASME, ASM, IEEE, INPO, EPRI, etc. will be searched to evaluate the system monitoring techniques as well as testing programs.

Task (2): Current Maintenance Practices Assessment

An industry survey will be conducted to assimilate various plant maintenance practices and procedures in relation to the subject system. The work performed in Task 7 of Phase I and the USNRC Maintenance Program will be valuable here. Both corrective and preventive/predictive maintenance programs will be reviewed. Other relevant programs to be reviewed include the plant maintenance management program, human reliability, training, QA/QC, and spare parts. Based on the review of the above, the advantages and deficiencies in the current industry practices will be assessed.

Task (3): Assessment of System Testing and Condition Monitoring

In this task BNL will review and analyze current NPP practices for each selected system in the area of system testing and condition monitoring. This will concentrate on integrated system functional testing but must also include testing of key components. Included in the review will be preoperational testing, inservice testing, surveillance testing, condition monitoring, inspection practices, and training.

Task (4): System Operation Evaluation

This task will collect and assess NPP operating techniques and procedures associated with the selected system. Included in the review will be normal and emergency operating procedure, actual operational methods, and associated training and qualification of the plant staff. For a normally operating system, the method of system operation can have a significant effect on how the system ages and how it will perform under the stress of an accident situation.

NPAR component aging studies revealed that frequent starts and stops, sudden valve closures, and many other operational related activities accelerate the aging degradations of both mechanical and electrical components in a system. For example, too much switching causes electrical surges and sparks on the contact surfaces, quick starts on motors overheat insulation causing accelerated aging of the insulating polymers, and thermal transients could increase the potential for crack growth in piping leading to eventual pressure boundary failures.Therefore, this task will review various plant component start up and operating procedures for the subject system.

The product of this task will be a list of the current, system specific, operating practices, their impact on equipment aging and recommendations for future practices.

Task (5): Integration of Tasks 1-4

Tasks 1-4 products will be integrated to establish the present state-of-the art in regulation and guidance, plant inspection, surveillance, monitoring, and maintenance programs, and system operating procedures. This section will discuss each of the above activities and develop a matrix to illustrate various procedures. The task will identify all high and low points of each program and their suitability for the system operational readiness.

4.2 Phase II Tasks

Task (1): Plant Risk Assessment

Using the PRA model developed for the specific plant, an importance measure study will be conducted to identify the critical components based on plant risk. The study will predict the probability of system failure as a function of system age in systems composed of components having diverse time-dependent hazard rates. The output of this study completed mainly in Phase I, will be the point or instantaneous system unavailability and its uncertainty as a function of time. The BNL-developed FRANTIC II code, and/or the MIT modification in conjunction with other codes will be used to propagate the uncertainties in the component hazard rates through the system models to determine the overall uncertainty in the system unavailability.

Reliability techniques as developed in the PETS program, will be used to optimize the AOTs and STIs for components under tech spec requirements. This will aid in reducing unnecessarily frequent tests on equipment as required by the present plant surveillance programs.

Task (2): Recommended Practices

This task is the culmination of all the work performed in Phase I and Phase II studies. With the knowledge of this current industry wide age-related system problems and the mitigation programs in effect, recommendations will be provided to the nuclear industry, as well as the regulating agency for improving the system reliability, through enhanced maintenance practices and monitoring techniques which will be focussed on the most risk sensitive areas of a system.

Task (3): Application to Plant Life Extension

The age-dependent plant risk analysis provides the basis for evaluating requests for life extension. It is imperative that the plant risk be maintained at or near the risk level that was the basis for the original licensing action. When the licensee submits a plan for life extension, ALEAP analysis will serve as a basis for considering and evaluating how the aging effects are managed for the specific systems under study. This may be done by equipment replacement and/or installing incipient failure detection devices. The ALEAP model would be re-evaluated, including these anti-aging provisions, but with consideration that the age arresting devices and procedures may fail. After suitable analysis if it is found that the risk is within NRC requirements, recommendations could be made regarding the specific systems under study being suitable for extended life.

5. UTILIZATION OF RESEARCH RESULTS

5.1 Phase I and II Results

The system level aging assessment will provide a better understanding of the system aging characteristics under normal, as well as accident and transient conditions. The study will prioritize critical components for maintenance and monitoring activities (both with and without aging considerations). Techniques to predict the expected life of certain equipment subcomponents will be established to assess the operational readiness of the system. Following is the list of some uses for research results:

- To support the NRC in review, development and inspection of maintenance and surveillance programs.
- To support the NRC to monitor and inspect systems.
- To identify failure modes, causes, and mechanisms associated with a particular system under study, and to identify the dominant modes affecting the system availability.
- To identify system and component level functional indicators for monitoring system performance.
- To provide recommendations for updating rules and regulations, regulatory guides, industry standards, etc.
- To provide a technical basis for assessing life extension issues for NPPs.
- To aid in evaluating storage and "mothballing" issues.
- To determine the risk associated with aging of components, systems, and plants.
- To aid in the resolution of pertinent unresolved safety issues and Generic Issues.

5.2 Interim Results

The following items will be produced as interim results as the program proceeds:

- Products to help resolve generic issues associated with each of the systems under study. As an example Generic Issue 65 relates to one of the first systems selected for study, the component cooling water system. Also Unresolved Safety Issue A-45 "Shutdown Decay Heat Removal Requirements" relates to one of the other initial systems to be studied at BNL, namely BWR Residual Heat Removal Systems. Generic Issue C-9, "RHR Heat Exchanger Tube Failures also related to the RHR system.

- System Inspection Guidance for I&E/Regional Offices to include system failure modes, failure causes, effects on the plant of failures, aging and service wear effects, system functional indicators, mitigation and detection techniques, recommendations for testing, inspection, or surveillance methods, and methods to prevent or mitigate system failure.
- Information on a system basis to aid NRR in licensing decisions, related to system design variations, system failure modes, and effective failure mitigation techniques.
- Technical Specification insights related to LCO's, AOTs, STIs, and required surveillance testing.
- Input on system reliability insights to the Operational Safety Reliability Research Program.
- Input to the Accident Sequence Evaluation Program (ASEP).

6. RELATIONSHIP TO ONGOING WORK

The nature of this investigation is such that it may effectively draw from the work performed and being performed in many other NRC and industry programs. Conversely its successful execution will greatly assist other programs. The primary interactions are in the nature of root-cause determination, time dependence of failure rates, PRA plant modeling and completeness thereof, value-impact decisions relating safety and operability, plant management, actions to mitigate the effects of aging, and life extension analysis. Within BNL a seminar was held with participants from several related programs. Pertinent questions and answers generated are included in Appendix A. During the performance of this program, the project team will maintain contact with the NRC program offices of Research, NRR, and I&E in order to receive input as to how this program can serve their needs with interim products related to work completed.

As an example of one interface, this research project must take cognizance of the work to improve technical specifications which are now generally considered to be complex and difficult to implement and may adversely impact safety. The Surveillance Test Intervals (STIs) and Allowed Outage Times (AOTs) as specified by the Technical Specifications are not directly based on risk nor do they consider the possibly increasing risk of plant operation as the components age. Major single point passive failures (such as the pressure vessel, and to some extent, the piping) are not properly considered in PRAs but may control the criteria of plant life extension because of the costs and difficulty of their replacement. It may be that the STIs and the AOTs will need re-interpretation to include age considerations for possible inclusionin the Program for Evaluating Technical Specifications (PETS). The PETS program is designed to utilize risk-based techniques to establish a firmer basis for AOTs and STIs while also supporting other potential technical specification needs as indicated by TSIP. Table 6-1 summarizes interfaces with other NRC projects.

Table 6-1 Interfaces With Other NRC Projects

		Input to/from Programs
1.	Probabilistic Evaluation of Technical Specifications (PETS)	Input/Output-Applicable evaluation models and data, age consideration in surveillance intervals and allowed outage times.
2.	Nuclear Plant Aging Research (NPAR) (of which this document is a part)	Input-Identification of aging related parameters to be measured and trended and applicable monitoring techniques. Output-Equipment reliability data, measures of effective monitoring, strategy for using condition monitoring for managing aging.
3.	Root Causes of Component Failures (RCCF) - Materials - Stresses - Environment	Input-Identification of important aging parameters to be monitored, root cause failure data and alert levels. Output-Prioritizing root age-causes for degraded performance and analysing corrective actions.
4.	Technical Specifications Improvement Program (TSIP)	Input-Aging data and aging significance in tech specs. Output-Basis for performance-based tech specs by identifying impacts of surveillance requirements and allowed outage times on component performance and measuring age changes in reliability.
5.	Maintenance & Surveillance Program Plan (MSPP)	Input-Effective of age corrective action techniques and maintenance management approaches. Output-Evaluation of age mitigating activities. Measure of maintenance & surveillance effects on aging.

Table 6-1 (Cont'd)

Project	Input to/from Programs
6. Vendor Inspection Programs	Input/Output-Determination of which components and vendors are currently experiencing problems and a determination of which areas of systems and components age significantly and need inspection attention.
7. Quality Assurance Programs	Input-Identification of systems or components that have propensity for aging and hence, need aging-dependence assessment. Output-Guidelines for age considerations in reliability maintenance as part of Quality Assurance.
8. Human Factors/Reliability Programs	Input-Operational, test, and maintenance effectiveness, identification of human performance aids to arrest time dependent degradation of performance. Output-Identification of management, test, and maintenance needs to maintain high plant, system, and component performance.
9. Performance Indicator Program (PI)	Input-Possible indicators to use at the system level and methods for evaluating selected indicators to tell when alert levels have been reached. Output-Possible additional items at the system or plant level that could add to the PI program.
10. Research to Support NRC Inspection Prioritization	Input-Methods to prioritize safety importance of equipment with consideration for time dependent degradation. Output-Suggested methods to inspect performance-based regulations for maintaining licensed level of safety.

Table 6-1 (Cont'd)

Project	Input to/from Programs
11. Operational Safety Reliability Research (OSRR)	Input-Reliability techniques and methods useful for both aging assessments and ongoing condition monitoring. Output-Items susceptible to aging that should receive attention in OSRR. Methods to alleviate premature aging to be implemented in an OSRR type program.

7. REFERENCES

1. Golay, M.J. and Moinzadeh, J.H. "Extending the Life of Nuclear Power Plants: Technical and Initiation Issues," MIT-E686-003, April 1986.

2. IEEE Std. 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical & Electronic Engineers, New York, 1974.

3. Marriott, P.W. "Nuclear Plant Life Extension: A Comprehensive Approach to Plant Improvements," Trans. Am. Nucl. Soc., 46, 576-577, June 1984.

4. Drago, J.P. "The In-Plant Reliability Data Base for Nuclear Power Plant Components: Data Collection and Methodology Report," ORNL/TM-8271, July 1982.

5. ORNL "Aging Trends in Nuclear Power Plants," Oak Ridge National Laboratory Review, 16, No. 4, p. 76, 1983.

6. Marin, P.S., et al., 1983 "Reactor Plant License/Lifetime Extension," ANS TANSAO 50, p. 328, 1985.

7. Carfagno, S.P. and Gibson, R.J., "A Review of Equipment Aging Theory and Technology," EPRI-NP-1558, September 1980.

8. Simonen, E.P., "Predicting Irradiation Effects on Alloy Aging Kinetics," Proceedings of International Conference on Nuclear Power Plant Aging, ASM, July 8-12, 1985.

9. Vegnes, A, "Understanding of the Phenomena of Materials Degradation by Aging and Embrittlement -- Margin of Safety and Solutions to Limit or Avoid Such Degradation", ASM op.cit pg. 165, 1985.

10. Sanoh, J. et al, 1985, "Irradiation Embrittlement Behavior of Some Japanese Reactor Vessel Steels," ASM op cit pg 169, 1985.

11. Hinton, B.M., "Plant Life Extension Program for Nuclear Piping System," ASM op cit pg. 253, 1985.

12. Moelling, D.S. and Gallucci, R.H.V., "Probabilistic Analysis of Integranular Stress Corrosion Crack Growth in BWR Recirculation Piping," ASM op cit pg 261, 1985.

13. Gordon, B.M. and Gordon, G.M., "Mitigation of Pipe Cracking for BWR Plant Life Extension," ASM op cit pg. 287, 1985.

14. Horak, J.A., et al, "Review of Effects of Long-Term Aging on the Mechanical Properties and Microstructures of Types 304 and 316 Stainless Steel," ASM op cit pg. 301, 1985.

15. Spond, D.F. et al., "Degradation and Replacement of Main Feedwater Nozzle Spray Heads in Two Once-Through Steam Generators," ASM op cit pg. 283, 1985.

16. Odette, G. et al, "Irradiation Effects on Aging of Pressure Vessel Steels" ASM op cit pg. 375, 1985.

17. Banford, W.H. et al, "Thermal Aging Issues in Nuclear Power Plant Piping Integrity," ASM, op cit pg. 370 1985.

18. Server, W.L., "Life Extension of Components with High Cumulative Fatigue Usage," ASM op cit pg. 375 1985.

19. Bush, S.H. et al, "Aging and Service Wear of Hydraulic Mechanical Snubbers Used on Safety-Related Piping and Components of Nuclear Power Plants," NUREG/CR-4279, February 1986.

20. Duframe, K. et al, "Wear Measurements of Nuclear Power Plant Components," EPRI NP-3444, May 1984.

21. Vause, J.W. et al, "Aging of Nuclear Station Diesel Generators: Evaluation of Operation and Expert Experience," NUREG/CR-4590, April 1986.

22. Dingee, D.D. & Johnson, Jr., A.B., "Aging and Service Wear of Diesel Engines Used for Emergency Power at Nuclear Power Stations," ASM op cit pg. 367, 1985.

23. Higgins, J.C. & Subudhi, M., "A Review of Emergency Diesel Generator Performance at Nuclear Power Plants," NUREG/CR-4440, 1985.

24. Vesely, W.E., & DeMoss, C.M., "Evaluation of Diesel Unavailability and Risk Effective Surveillance Test Intervals," SAIC report for BNL PETS Program, April 1986.

25. Subudhi, M. et al, "Operating Experience and Aging-Seismic Assessment of Electric Motors," NUREG/CR-4156, June 1986.

26. Subudhi, M. et al, "Recommended Maintenance Practice for Improving Motor Reliability in Reactor Power Plants," BNL FIN A-3270, September 1985.

27. Taylor, J.H. et al, "Seismic Endurance Tests of Naturally Aged Small Electric Motors," BNL Technical Report A-3270-11-85, November 1985.

28. Subudhi, et al, 1985 (Aging/Seismic).

29. Sugarman, A.C., "Aging-Seismic Correlation Study on Class 1E Equipment," NUREG/CR-3808, September 1984.

30. Bonzon, L.L. & Hente, D.B., "Test Series 1: Seismic-Fragility Tests of Naturally-Aged Class 1E Gould No. CA-2250 Battery Cables," NUREG/CR-3923, September 1984.

31. Greenstreet, W.L.et al, "Aging and Service Wear of Check Valves Used in Engineered Safety-Feature Systems of Nuclear Power Plants," NUREG/CR-4302, Vol. 1, December 1985.

32. Subudhi, M. & Taylor, J.H., "Indian Point 2 Reactor Coolant Pump Seal Evaluation," BNL A-3270/87719-7-86, 1985.

33. Toman, G.J. et al, "The Interaction Effects of Relay and Circuit Breaker Aging in a Safety-Related System," NUREG/CR-4715, July 1986.

34. Gunther, W.E. et al, "Operating Experience and Aging Seismic Assessment of Battery Chargers and Inverters," NUREG/CR-4564, June 1986.

35. Stuetzer, O., "Status Report: Correlation of Electrical Cable Failure with Mechanical Degradation," NUREG/CR-3263, April 1984.

36. USNRC, "Nuclear Plant Aging Research (NPAR) Program Plan," NUREG-1144, July 1985.

37. Fullwood, R, "Incipient Failure Detection Methods for Nuclear Power Plants: Reality and Possibilities," BNL Report in printing, 1986.

38. Sliter, G.E. & Carey, J.J., "EPRI Research on Component Aging and Nuclear Plant Life Extension," ASM op cit pg. 105, 1985.

39. Plumstead, W.C. & Cady, D.R., "NDE Application in Power Plant Life Assessment Studies," ASM op cit pg. 53, 1985.

40. Engl, G. & Figlhubur, D., "Use of Nondestructive Evaluation (NDE) Methods to Improve Power Plant Availability Factors," ASM op cit. pg. 57, 1985.

41. Weber, R.M., "Use of Nondestructive Evaluation Methods to Improve Power Plant Availability," ASM op cit. pg. 61, 1985.

42. Rose, J.A. et al, "Survey of Aged Power Plant Facilities," NUREG/CR-3819, June 1985.

43. Leverenz, et al, "Using PRA to Rank Importance of Aging on NPP Components," ASM op cit. pg. 639, 1985.

44. Silver, M. and Vasudevan, R., "Pilot Assessment: Impact of Aging on the Seismic Performance of Selected Equipment Types," EQE Report for BNL Contract 223951-S, December, 1985.

45. Davis, T., et al, "Importance Ranking Based on Aging Consideration of Components Included in Probabilistic Risk Assessments," NUREG/CR-4144, April 1985.

APPENDIX A

AGING QUESTIONS AND ANSWERS

The following questions and answers arose during a seminar at Brookhaven National Laboratory when ALEAP was introduced to a constructively critical audience. They are presented for further understanding of the program.

Q. How do you decide how old is old enough?

A. A very difficult question because with replacement of aged components, operation and safety can be sustained. It seems that this is eventually an economic problem concerning the replacing of equipment to maintain the level of safety at or near the level on which the plant was originally licensed. Also related is the question of when to replace components.

A hybrid approach is probably necessary combining operating experience, expert opinion, in depth engineering studies, inspection, condition monitoring, vendor input, trending, etc. This study fills part of the need.

Q. How much can the age-caused risk rise before action must be taken to mitigate the aging effects?

A. This is basically the question of how safe is safe enough. We do not feel that it is our mission to establish these safety criteria but to provide the NRC with the necessary tools on which to base the regulations.

Q. How many models will be needed to represent the nuclear industry in the US?

A. This is unknown and depends on the system. There are something like 60 models of the Service Water System to represent the industry used in ASEP, but only a few for more standard systems. One of the tasks will be to establish an aging correlation matrix to determine permissible plant grouping without serious loss of specificity.

Q. How do you propose to reduce the plant complexity to a do-able problem?

A. We propose to prioritize the plant systems and components according to the safety significance of the aging using PRA and fault tree techniques.

Q. How do you measure this safety significance?

A. We propose to use a PRA model of the plant and to rank the systems and in turn their components according to their importance to the plant safety at any given time. We will place the primary effort on those components that are the most important and address the ones of lesser importance on a resources availability basis.

Q. What measure of importance will be used?

A. This is not settled but for the time being it appears that the Age Importance measure will be defined as:
Frequence of Age-Caused Failure x Change in Risk if the component under consideration fails.

Q. Is Age Importance independent of time?

A. No. It must be evaluated for various time periods.

Q. Will this change the prioritization of components for aging investigation?

A. Possibly. If the assumption that the components that have the highest aging failure rate are also those that are the most important to aging is true, then the initial prioritization should be maintained. One must realize that the Age Importance measure is complex. The frequency of age-caused failure may be time dependent but so is the change in risk if it fails because the components involved in this latter quantity are also changing.

Q. How can you be sure that the components most significant to aging were included in the PRA?

A. A reexamination of the PRA must be done to assure the consideration of things such as piping and the pressure vessel until it can be shown that they are not significant.

Q. How will common cause dependencies be addressed?

A. They will be addressed in the PRA model. It may be necessary to perform additional analyses to characterize the operating environment of the components at least in so far as this contributes to aging. There are several codes available for this, such as INEL's COMCAN, SETS, and the WAM series.

Q To what extent can industry-wide data be used?

A. One of the tasks is to examine its applicability. It may be so heterogeneous that an age dependence cannot be extracted. Knowing that a component is failing at some rate as is usually the best that can be extracted from say, NPRDS but aging analysis requires knowledge regarding how long it was in service, the test and maintenance practices and what was the root cause of failure.

Q. Why do you need to know the root cause?

A. To determine if the cause of failure was age-dependent. This may be fairly broad and include age dependence in human error.

Q. How do you factor trigger events (accident initiators) into the study?

A. This must be by analyses, such as PRA, and special review and extension of the failure data bases.

Q. What about new initiators which are not currently in PRAs, but may be important at a later lagged/time?

A. One can never be 100% complete, but every effort will be made to identify new initiators creeping into the failure data bases. Also, analyses can identify potential new initiators for aged systems.

Q. What about cutsets (for PRA system fault trees) that are not now dominant but may become dominant as the systems age? Or new failure modes due to aging that are not currently included in fault trees and hence for which there would be no cutsets?

A. Must be careful not to truncate low probability cutsets. Also one may need to modify fault trees with new failure modes (due to aging) and hence, generate new cutsets.

Q. What information do we want from selected plants to be visited?

A. Information such as:

Maintenance History	Technical Specifications
Failure Data	Drawings
P&IDs	ISI/IST Information
Procedures	Reliability Data
Test Information	Spatial Layouts
PRA Information	Materials Used
Stresses	Operating Environment

APPENDIX B

PRELIMINARY PRIORITIZATION OF INDIAN POINT 3 CCW SYSTEM

B.1 Safety Ranking of Components

As a demonstration of the methodology for prioritizing the safety significance of component aging, the process begins with the CCW system cutsets. Table B-1 presents the component cooling water first and second order cutsets from the Indian Point Probabilistic Safety Study (IPPSS). The rule that has been followed is that the total importance of a component is the sum of its importances in the accident sequences in which it appears. Table B-2 presents a factored grouping of these added cutsets with the leading term being the component of interest. This component identifier multiplies the sum of additional terms that contribute to its total importance. Thus a single cutset appears alone and by itself as does for example, UPPLEAKS. Some terms are paired with single terms such as UTK0031L*UTK0032L so there is one listing for the first term and another for the second term. This table provides the algebraic representation of the Inspection Importance of each of the components. To obtain a numerical value requires the substitution of the appropriate non-aged failure rate data. This is provided in Table B-3 in which the component is identified as before, the non-aged failure rate from the IPPSS is presented and a brief description of the component and the type of failure. Using the failure rates in this table and the algebraic representations of the Inspection Importance presented in Table B-2, the individual component importances may be calculated and ordered by descending importance. TAble B-4 provides such a presentation. It will be noted that the Service Water is the most important "component" of the CCW system because it must remove the heat taken up by the CCW. This study will address the interface only, and not examine the details of how/why service water fails. Next in importance is switchgear bus 2A because its failure is a common cause failure of the CCW. Least in importance are the surge tanks because of the low likelihood of failure and their redundancy.

Table B-1 CCW System Cutsets

First Order
1 UPPLEAKS
2 TXV31--C
3 TSW1NOFL

Second Order
1 UTK0031L UTK0032L
2 UXV759AC UXV759BC
3 UHE0031L UXV759BC
4 UXV765AC UXV759BC
5 TXV034AC UXV759BC
6 TXV035AC UXV759BC
7 UXV759AC UHE0032L
8 UHE0031L UHE0032L
9 UXV765AC UHE0032L
10 TXV034AC UHE0032L
11 TXV035AC UHE0032L
12 UXV759AC UXV765BC
13 UHE0031L UXV765BC
14 UXV765AC UXV765BC
15 TXV034AC UXV765BC
16 TXV035AC UXV765BC
17 UXV759AC TXV034BC
18 UHE0031L TXV034BC
19 UXV765AC TXV034BC
20 TXV034AC TXV034BC
21 TXV035AC TXV034BC
22 UXV759AC TXV035BC
23 UHE0031L TXV035BC
24 UXV765AC TXV035BC
25 TXV034AC TXV035BC
26 TXV035AC TXV035BC
27 TXV33--C UXV759BC
28 TXV33-1C UXV759BC
29 TXV33--C UHE0032L
30 TXV33-1C UHE0032L
31 TXV33--C UXV765BC
32 TXV33-1C UXV765BC
33 TXV33--C TXV034BC
34 TXV33-1C TXV034BC
35 TXV33--C TXV035BC
36 TXV33-1C TXV035BC
37 UES-35AC 4BS-333C
38 4BS-331D 4BS-333C
39 UXV760AC 4BS-333D
40 UM00031S 4BS-333C
41 UXV762AC 4BS-333D
42 UPM0031S 4BS-333D
43 UCV761AQ 4BS-333D
44 UCC0031F 4BS-333C

Table B-2

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UPPLEAKS TXV31--C TSW1NOFL UTK0031L*UTK0032L UTK0032L*UTK0031L
UXV759AC*(UXV759BC+UHE0032L+UXV765BC+TXV034BC+TXV035BC)
UXV759BC*(UXV759AC+UHE0031L+UXV765AC+TXV034AC+TXV035AC+TXV33--C+TXV33-1C)
UHE0031L*(UXV759BC+TXV035BC)
UXV765AC*(UXV759BC+UHE0032L+UXV765BC+TXV034BC+TXV035BC)
TXV034AC*(UXV759BC+UHE0032L+UXV765BC+TXV034BC+TXV035BC)
TXV035AC*(UXV759BC+UHE0032L+UXV765BC+TXV034BC+TXV035BC)
UHE0032L*(UXV759AC+UHE0031L+UXV765AC+TXV034AC+TXV035AC+TXV33--C+TXV33-1C)
UXV765BC*(UXV759AC+TXV034AC+UXV765AC+TXV034AC+TXV035AC+TXV33--C+TXV33-1C)
TXV034BC*(UXV759AC+UHE0031L+UXV765AC+TXV034AC+TXV035AC+TXV33--C+TXV33-1C)
TXV035BC*(UXV759AC+UHE0031L+UXV765AC+TXV034AC+TXV035AC+TXV33--C+TXV33-1C)
TXV33--C*(UXV759BC+UXV759BC+UHE0032L+UXV765BC+TXV034BC+TXV035BC)
TXV33-1C*(UXV759BC+UHE0032L+UXV765BC+TXV034BC+TXV035BC+UXV759BC)
UES-35AC*4BS-333C 4BS-333C*(UES-35AC+4BS-331D+UM00031S+UCC0031F)
4BS-333D*(UXV760AC+UXV762AC+UPM0031S+UCV761AQ) UXV760AC*4BS-333D
UM00031S*4BS-333C UXV762AC*4BS-333D UPM0031S*4BS-333D UCV761AQ*4BS-333D
UCC0031F*4BS-333C 4BS-331D*4BS-333C
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Table B-3

Component Identifier, Non-Aged Failure Rate and Description of the Component and Failure Mode

Identifier	Failure Rate	Component Description and Failure Mode
UPPLEAKS	1.46E-8/H	17 major sections of CCS piping
TXV31--C	9.15E-8/H	SW supply vlv 31, transfers closed
TSW1NOFLa	4.2E-5/H	No flow from SW supply header (conventional)
UTK0031L	8.6E-10/H	CC surge tank 31 leak or rupture
UTK0032L	8.6E-10/H	CC surge tank 32 leak or rupture
UXV759AC	9.15E-8/H	Heat exch 31 inlet valve, transfers closed
UXV759BC	9.15E-8/H	Heat exch 32 inlet valve, transfers closed
UHE0031L	9.73E-7/H	CC heat exch 31, loss of cooling cap. (leak or rupture)
UXV765AC	9.15E-8/H	Heat exch 31 outlet valve, transfers closed
TXV034AC	9.15E-8/H	SW inlet to heat exchanger 31, trans. closed
TXV035AC	9.15E-8/H	SW outlet to heat exchanger 31, trans clsd
UHE0032L	9.73E-7/H	CC heat exch 31, loss of cooling cap. (leak or rupture)
UXV765BC	9.15E-8/H	Heat exch 32 outlet valve, transfers closed
TXV034BC	9.15E-8/H	SW inlet to heat exchanger 32, trans. closed
TXV035BC	9.15E-8/H	SW outlet to heat exchanger 31, trans clsd
TXV33--C	9.15E-8/H	SW supply vlv 33, transfers closed
TXV33-1C	9.15E-8/H	SW supply vlv 34, transfers closed
UES-35AC	Component does not appear in fault trees - no data	
4BS-333C	5.24E-5/H	No control pwr at switchgear bus 2A
4BS-332Db	5.24E-5/H	No control pwr at switchgear bus 2A
UXV760AC	9.15E-8/H	Pump 31 suction valve transfers closed
UM00031S	1.36E-3/D 3.26E-6/H	CC pump/mtr 31 does not start/run
UXV762AC	9.15E-8/H	Pump 31 discharge valve transfers closed
UPM0031S	1.36E-3/D 3.26E-6/H	CC pump/mtr 31 does not start/run
UCV761AQ	6.91E-5/H	Pump 31 discharge check vlv transfers closed
UCC0031F	This component was not found in fault trees - no data	
4BS-331D		No control pwr at switchgear bus 5A

Notes: a) IPPSS page if added gives 1.E-3/24 hrs.
b) Apparently a misprint. The fault trees show 4BS-332D which is assumed to be correct.

Table B-4

Identifier	Insp. Impt.	Component Description and Failure Mode
TSW1NOFL	3.02E-2	No flow from SW supply header (conventional)
4BS-333D	1.96E-3	No control pwr at switchgear bus 2A
UXV760AC	1.96E-3	Pump 31 suction valve transfers closed
UCV761AQ	1.84E-3	Pump 31 discharge checkvalve trans closed
4BS-333C	1.5E-4	No control pwr at switchgear bus 2A
UPM0031S	1.4E-4	CC pump/mtr 31 does not start/run
UM00031S	1.37E-4	CC pump/mtr 31 does not start/run
TXV31--C	6.58E-5	SW supply vlv 31, transfers closed
4BS-331D	1.06E-5	No control pwr at switchgear bus 5A
UPPLEAKS	1.05E-5	17 major sections of CCS piping
TXV035AC	2.6E-6	SW outlet to heat exchanger 31, trans clsd
UXV762AC	2.5E-6	Pump 31 discharge valve transfers closed
UHE0032L	3.3E-8	CC heat exch 31, loss of cooling cap. (leak or rupture)
UXV765BC	3.3E-8	Heat exch 32 outlet valve, transfers closed
TXV034BC	3.3E-8	SW inlet to heat exchanger 32, trans. closed
TXV035BC	3.3E-8	SW outlet to heat exchanger 31, trans clsd
UXV759BC	3.0E-8	Heat exch 32 inlet valve, transfers closed
UXV765AC	2.64E-8	Heat exch 31 outlet valve, transfers closed
TXV034AC	2.6E-8	SW inlet to heat exchanger 31, trans. closed
UXV759AC	2.2E-8	Heat exch 31 inlet valve, transfers closed
TXV33--C	2.2E-8	SW supply vlv 33, transfers closed
TXV33-1C	2.2E-8	SW supply vlv 34, transfers closed
UHE0031L	9.24E-9	CC heat exch 31, loss of cooling cap. (leak or rupture)
UTK0031L	2.26E-13	CC surge tank 31 leak or rupture
UTK0032L	2.26E-13	CC surge tank 32 leak or rupture

APPENDIX C

SAMPLE SYSTEM SURVEY

As discussed in Phase I, Task 3 support systems such as Component Cooling Water (CCW) vary considerably between NPPs and hence the design variations must be surveyed and analyzed as a part of the overall system analysis. Below is an example of a system survey for one plants CCW system.

Component Cooling Water (CCW) System Summary

Plant: Recent 2 Unit Westinghouse PWR   System Designer: Sargent & Lundy

Info Source: FSAR, System Description, P & IDs

Pumps:

Number: 5   Head: TDG = 250'

Flow Rate: 4800 gpm   Motor Horsepower: 460

Elec. Source:5 pumps on 4 - 4160 volt, ESF, busers
One motor each on Bus # 141, 142, 241, 242
The fifth motor (pump-0) on any of the 4 buses.

Heat Exchangers (Hx): Number: 3

Surge Tanks: Number: 2

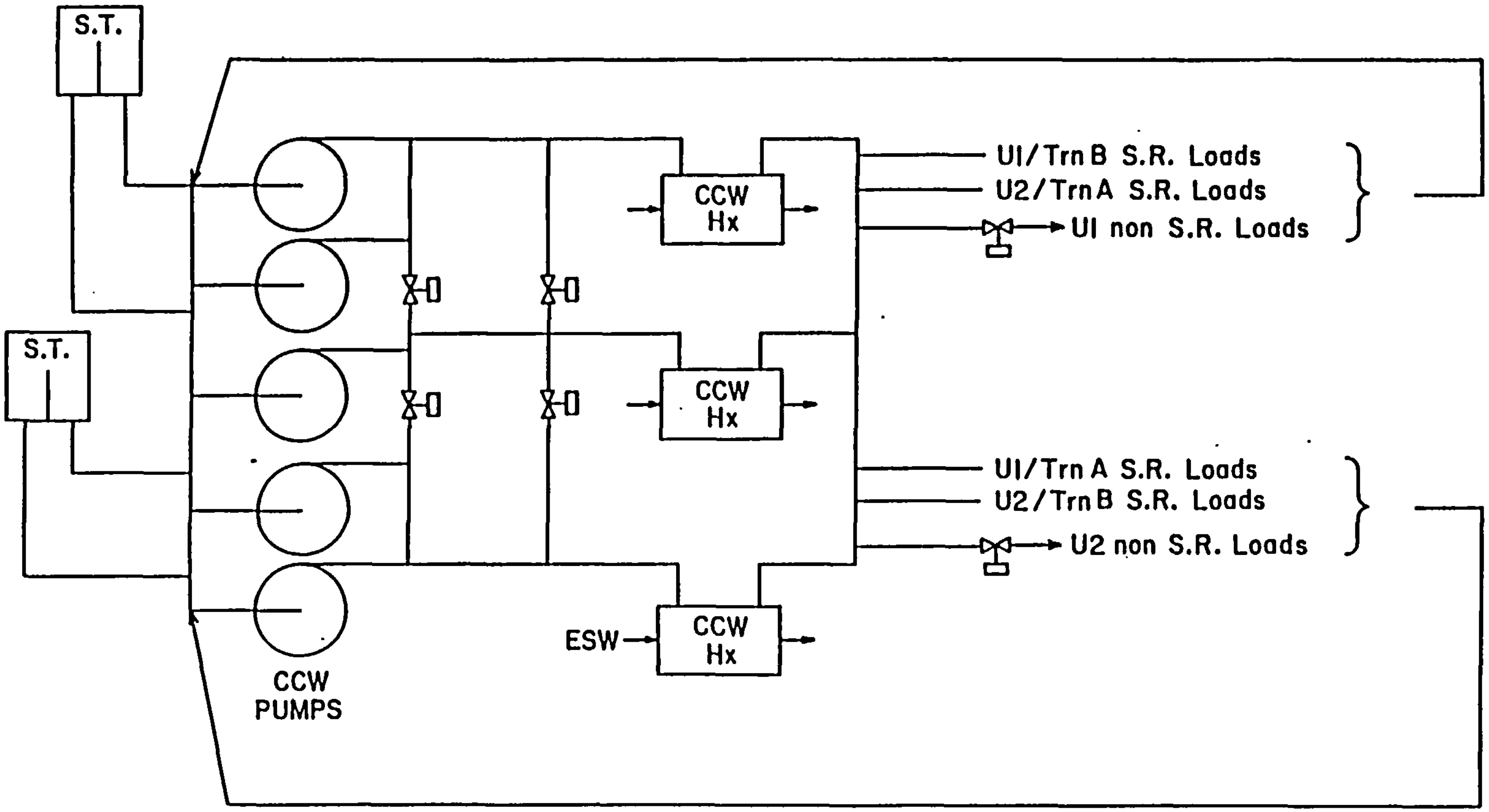
Cooling by: Essential Service Water (ESW)

Loads: RHR pumps, RHR HXs, Rx Coolant Pump Motor and Thermal Barrier, Positive Displacement Charging Pump, Spent Fuel Pool Hx, Letdown Hx, Seal Water Hx, Excess Letdown Hx, Miscellaneous Loads.

Notes: CCW is a shared, normally cross-connected, system between Units 1 & 2.

	Indication	Alarms	Interlocks
Instrumentation:	CCW pump suction temp. CCW HX outlet temp. CCW pump disch. press. CCW pump suction flow RCP Flow & Misc. Flows	Same as ind.	Auto pump start on low discharge pressure.

SIMPLIFIED CCW SYSTEM



KEY

S.T. = Surge Tank
CCW = Component Cooling Water
Hx = Heat Exchanges
ESW = Essential Service Water

Ul = Unit One
U2 = Unit Two
Trn = Train
S.R. = Safety Related