Interim Report

NRC RELIABILITY PROGRAM PLAN

VOLUME II
APPENDICES

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APPENDIX A

QUALITY ASSURANCE AND RELIABILITY DEFINITIONS

This appendix contains terms and definitions identified through a review of documentation and information compiled during the study. Many of the terms and definitions are based on ANSI/IEEE Standard-761 which defines reliability, availability and productivity terms for electric power generation systems.

APPENDIX A

QUALITY ASSURANCE AND RELIABILITY DEFINITIONS

Accelerated Test - A test in which the applied stress level is chosen to exceed the level stated in the reference conditions to shorten the time required to observe the stress response of the item or magnify the response in a ggiven time. To be valid, an accelerated test must not alter the basic modes and/or mechanisms of failure of their relative prevalence.

Acceleration Factor - The ratio between the times necessary to obtain a stated proportion of failures for two different sets of stress conditions involving the same failure modes and/or mechanisms.

Adequacy - Sufficient generating capability to meet the aggregate peak electric loads (MW) and energy requirements (MWh/h) of all customers at all times.

Availability - The probability that a material, component, equipment, system, or process is in its intended functional condition at a given time and therefore is either in use or capable of being used under a stated environment.

Availability (Equivalent) - The percentage of time in a period that gross maximum generation could be produced if limited only by outages and unit and seasonal derating.

Availability (Operating) - The percentage of time in a period that the system, process, or facility is operating or is available to operate (ready status). This measure ignores partial outages, i.e., if the system is producting any product at all, it is considered to be "available."

Capacity - The net power output for which a generating unit or station is rated.

Capacity, Gross Maximum - The maximum capacity that a unit can produce over a specified period of time.

Capacity, Gross Dependable - The gross maximum capacity modified for ambient limitations for a specified period of time, such as a month or a season.

Capacity, Gross Available - The gross dependable capacity modified for equipment limitation at any time.

Capacity Factor - A percentage calculated from the ratio of product actually produced in a period to the product that would be produced if the process system or facility operated at full rated capacity for the period.

Confidence Level - Statistical boundaries limiting an estimate with a specified risk.

Configuration Management - A technical and administrative process used to identify, control, and account for engineering documents describing the functional and physical characteristics of components, equipment, systems, or a process. It is also used to track and control hardware to conform to the documentation.

Corrective Maintenance - All unscheduled inspection, testing, or repair activities performed on equipment, following its failure, for the purpose of restoring the equipment to satisfactory operating condition.

Critical Item - A procedure, material, component, or item of equipment whose failure could significantly affect safety, performance, environment, schedule, or cost.

Debugging (Burn-In) - A process of shaking down each item of finished equipment that is performed prior to placing the item in use. During this debugging period, weak system elements are expected to fail and be replaced by elements of normal quality (statistically) that are not subject to similar early failure. The debugging process may involve exposure to all field operational stresses. The debugging process is not, however, intended to detect inherent weaknesses in system design, which should have been eliminated in the preproduction stages by appropriate techniques. The debugging process eliminates the parts subject to infant mortality.

Demonstration Plant - An RD&D project designed to demonstrate and validate economic, environmental, and productive capacity of a near-commercial size plant by integrating and operating a single modular unit using commercial-size components.

Derating, Seasonal - The difference between gross maximum capacity and gross dependable capacity.

Derating, Unit - The difference between gross dependable capacity and gross available capacity.

Design Life - The expected time or number of cycles, based on the design of the item, during which the item remains operationally effective and economically useful before wearing out.

Design Reviews - Meetings held during the design process to critically examine the product design, configuration, design documentation, test program planning, and test data.

Design Review, Critical (CDR) - A formal customer review of all accomplishments during detailed design. This may entail review of prereleased detailed design documentation; e.g., drawings and specifications, analytical and experimental verification data, long lead item procurement list, bid package plan, siting and environmental impacts, final test and evaluation plan, configuration, and change control procedures.

Design Review, Preliminary (PDR) - A formal customer review of process analyses and flow, reaction rates, operating parameters, including identified layout arrangement of equipment/systems, performance requirements, specifications of long lead items, and test plans.

Destructive Testing - Testing of any nature that may materially affect the life expectancy of the item tested, whether or not failures occur during the test.

Failure - The cessation of the ability of a system or any of its elements to perform a specified function or functions.

Failure Analysis - The study of a specific failure to determine the failure mode, mechanisms, and/or the circumstances that caused the failure.

Failure Effect - A description of the consequence of the failure in terms of operating or performance characteristics; e.g., shutdown, loss in efficiency, safety hazard.

Failure Mechanism - The physical process or occurrence that caused a failure (e.g., stress corrosion cracking, operator error, equipment malfunction, relay contacts welded by overload, and bearing frozen by contamination with foreign material).

Failure Mode - The observed local result of a failure, e.g., leak, loss of control, false output.

Failure Modes and Effects Analysis (FMEA) - Identification and documentation of each significant failure mode of each item and the impact of the occurrence of that mode of failure on the component, other components, and the overall operation of the system.

Failure (Noncurtailing) - A component failure that occurs with no effect on the output of the plant.

Failure Rate (Failure Per Unit of Time) - The number of failures per unit of time in a specified time interval.

Failure Reporting and Corrective Action - A systematic and comprehensive method of reporting failures and a means for implementing the corrective maintenance indicated by these failures.

Fault Tree Analysis - A method for relating a process or system failure to equipment, component, or materials failure modes using fault trees. A fault tree is a model that graphically and logically represents the various combinations of possible events, fault and normal, occurring in a process or system that leads to the top event. Process or system elements may include hardware, software, and human and environmental factors.

Forced Outage Rate - The ratio of forced outage hours to operating hours, plus forced outage hours.

Functional Configuration Audit - A formal examination of test data, prior to acceptance, to verify compliance of measured performance with specification requirements.

Functional Test - A test that directly or indirectly measures a specific function of equipment or a component.

Hazard - Any real or potential condition that can cause injury or death to personnel or damage to or loss of equipment of property.

Life-Cycle Cost Analysis - A function whose objective is to optimize the economics resulting from costs expended for design, construction, operation, and maintenance of equipment, a component, a system, or a process. LCC analyses are significant for:

- Assisting engineering in design trade-offs by providing a baseline of total life-cycle costs for all major design alternatives
- Providing a basis for determining the least cost involved in other major project alternative (e.g., maintenance concept development, planning system operation and support activities, and maintenance planning).

Load Factor - The ratio of the actual energy supplied during a designated period to energy that would have been supplied if the peak load were to exist throughout the designated period.

Loss-Of-Load Probability (LOLP) - The proportion of time that the generation available is unable to meet the system load (kilowatt). The loss-of-load probability is normally expressed in terms of days when the load is not met in the years studied, e.g., an LOLP of one day in 10 years means the load is not met one day in a period of 10 years.

Maintainability - A characteristic of design and installation that is expressed as the probability that an item can be restored to operation within a specified period of time when maintenance is performed in accordance with prescribed procedures and resources.

Maintenance - All actions necessary for retaining an item in a specified condition before failure or breakdown (preventive maintenance) or the process of restoring an item to return it to a workable condition (corrective maintenance).

Maintenance Concept - A description of the planned general scheme for maintenance and support covering replacement or repair philosophies, personnel factors, maintenance time, schedules, special tools, materials, supplies, spares, and other resources required to perform either corrective or preventive maintenance.

Maintenance Engineering Analyses - An analytical process in which the quantitative requirements, support resources, cost, operational objectives, and safety considerations that effect each preventive and anticipated corrective maintenance action are estimated and documented.

Mean Time Between Failures (MTBF) - Total operating time (frequently stated in hours) divided by the total number of failures.

Mean Time To Outage (MTTO) - Operating time divided by the number of outages experienced. This measure can be calculated for full, partial, or all outages (i.e., planned, forced) and is most applicable to mature technologies.

Mean Time To Resotre (MTTR) - Average time to restore the system, process, or facility to full operability after an outage (full, partial, or all). This measure includes all maintenance and delay time encountered in restoring a system, e.g., waiting for parts. It can be used for identifying problems early in the maintenance program, the operational procedures, and the system design. It also includes time used for operator-initiated restoration activites (e.g., restart actions), which may restore the system in lieu of performing a maintenance action.

lean Time To Repair (MTTR) - Average time to repair (or replace) a failed item during corrective maintenance. This measure excludes time waiting for parts, travel time, and other time not directly associated with the performance of the corrective maintenance activity. It can be used for identifying problems early in the maintenance program and/or system design characteristics affecting maintenance.

lean Maintenance Man-Hours - Average total maintenance man-hours required to perform preventive maintenance (servicing) and corrective maintenance (repairs or replacements of failed items). This measure is important in evaluating required maintenance staffing and projecting future maintenance costs.

londestructive Testing - A test that is neither functional nor potentially destructive. It is performed to establish acceptability, e.g., X-ray analysis, leak tests, ultrasonic tests, etc.

n-Stream Time - The percentage of time in a period that the system, process, or facility is producing products equivalent to "operating availability."

utage, forced - The failure of a system resulting a loss of all or part of the output. A full outage results in complete loss of output; a partial outage results in degraded system output.

utage, Planned - The period a unit is unavailable due to inspection, testing, nuclear refueling, or overhaul. A planned outage is scheduled well in advance and is of a predetermined duration.

ercentage Reserve - The margin of installed capacity in excess of the expected peak load.

erformance Assurance - A method for the systematic treatment of reliability, maintainability, availability, life-cycle cost, standardization, configuration management, and quality assurance in the design, construction, and operation of a system.

Performance Indices - Completely describe the performance of electrical-power operating unit. Included are capacity factor, availability (operating and equivalent), and outage rates (forced and planned).

Pilot Plant - An R&D project designed to establish integrated process feasibility by combining commercial type (not commercial size) components into a small model plant to test and evaluate the critical parameters of scale-up. It also is used to acquire engineering data needed to assess economic feasibility and design a larger near-commercial size plant.

Plant - The aggregate of major systems, personnel, procedures, and practices that perform the collective total functions of a process (e.g., coal liquefaction).

Plant Operability - The overall cost-effective plant generation needed to produce a required quantity of acceptable products at a predictable rate and at an acceptable level of reliability.

Preventive Maintenance - Actions performed in an attempt to keep an item in working condition and prevent failures or degradation of performance characteristics by planned (usually scheduled) servicing, replacement, overhaul, etc.

Quality Assurance - The system of engineering activities that assures quality by performing and preparing implementation documents for quality control. It includes analyzing all quality-related considerations for the development, implementation, and continuing evaluation of a quality control system.

Quality Control - The system of inspection and testing activities that are performed, documented, and used to measure, monitor, and control quality as well as to initiate corrective and/or preventive action in controlling selected characteristics of an item. It also performs the acceptance or rejection function at key points in the evolution and/or use of a product and implements the quality control system developed by quality assurance.

Redundancy - The existence of more than one means for accomplishing a given task, where all means must fail before there is an overall failure to the system.

Redundancy, Parallel - The existence of two systems working at the same time to accomplish the task, where either system can handle the job itself in case the other system fails.

Redundancy, Standby - The existence of an alternate means of accomplishing the task which is switched in by a malfunction sensing device when the primary system fails.

Reliability - The probability that an item or process performs its intended function for a specified time interval under a stated environment.

Reliability, Dynamic - The ability to withstand a sudden outage in its first few seconds or minutes without causing additional loss of facilities (i.e., preventing a cascading effect that may lead to widespread blackout).

Reliability, Inherent - The potential reliability present in an item's design.

Reliability, Operational - The assessed reliability of an item based on field data.

Reliability, Starting - The ratio of staring successes to total number of starting attempts.

Reliability, Steady-State - The system's ability to meet demand within specified voltage limits and the ratings of transmission lines during outages of some generating units and transmission lines.

Risk - The probability of occurrence of a specific deleterious consequence with a specific dimension, e.g., number of fatalities.

Security - System reliability in the steady-state and dynamic sense assuring actual operation, in contrast to its assessment (used by utility operating personnel).

Standard - A prescribed set of rules, conditions, or requirements established by standards setting bodies, concerning definition of terms, classification of components, specification of materials, performance or operations, delineation of procedures, or measurement of quality and quality in describing materials, products, systems, services or practices.

Service Life - The period of time during which a material, component, equipment, system, or process is expected to perform in a satisfactory manner under specified operational conditions prior to wear out or obsolescence and consequent removal from service.

Subsystem - A combination of personnel, equipment, procedures, and practices that performs a subgroup of functions within a system, e.g., "carbon burn-up cell," "hot gas cyclone," and "coal preheater."

System (Major) - A combination of personnel, equipment, procedures, and practices which performs a distinct group of functions within the plant, e.g., "utilities system," "coal preparation," "fluidized bed combustion system," "gas cleanup system".

System Safety Engineering - The activities indentified with the analysis of system design and operation for the timely identification and elimination of hazards. System safety activities closely parallel those of reliability to ensure that system safety is achieved early in the design phase and maintained throughout the system life cycle.

Trade-Off Analyses - Studies performed to optimize design in which interrelationships among performance, technical risk, cost, schedule, and safety are established and the effects of variations in these factors are determined.

Useful Life - The length of time an item operates with an acceptable failure rate.

APPENDIX B

SUMMARY OF UTILITY INTERVIEWS

This appendix presents a brief summary of the findings of the survey interviews which where conducted at the utility facilities. Those details considered proprietary sensiti~~on~~ve are not included in this appendix. The names of the persons interviewed are presented in Table 2-3 of Section 2.3.

APPENDIX B
SUMMARY OF UTILITY INTERVIEWS

COMMONWEALTH EDISON

Commonwealth Edison has developed a substantial reliability program that is oriented towards productivity improvement in the operating stations. This program was initiated five years ago with documentation and formal implementation beginning three years ago. The program has an initial goal of regaining 10% of lost operating hours. The methodology for achieving this goal involves identifying productivity problems, determination of causes and corrective actions, estimation of productivity improvement, monitoring implementation and estimation of cost effectiveness. Priorities for implementation of improvements are established in accordance with a "Top Twenty" list of problems in order of Nonoperating Hours caused.

The National Electric Research Councile (NERC) data format is used for recording outage data for 130 activities and components of a unit at a generating station. These data are given to NERC quarterly and processed annually into a Nonproductivity Report. The report provides a measurement of Unit Nonoperating Hours caused by each NERC component and activity.

The "Top Twenty" is selected based on those items whose Unit Nonoperation Hours, times the average annual fuel replacement cost, reveal the greatest cumulative costs. These economic measurements provide an upper bound for budgeting engineering analysis and retrofit.

The productivity improvement program, although lossely under the purview of reliability, incorporates very little actual reliability technology. Formal reliability analyses are not conducted on units, or subsystems of units. The reliability group contains no expertise in reliability engineering, but rather, the specialists within the engineering department. A program is under consideration to employ more reliability analyses in the productivity mprovement program. The nature of the reliability program anticipates the time hen all nuclear stations are operating, and little procurement or construction ctivity is present.

PACIFIC GAS & ELECTRIC COMPANY

Pacific Gas & Electric company has no formal reliability program at this time, but they are planning one for the future. Although there is no formal reliability program, many of the things being done, e.g., trend analysis, could be considered elements of such a program. Also, the economic incentive is there, in terms of providing increased plant availability.

In order to meet their requirements for data collection and analysis, they are in the process of designing a computerized maintenance following system, which will consist of a central computer with remote terminals. In addition to collecting failure and repair data, it will be used to perform forecasting, trend analyses, impact of design changes, failure predictions, reliability analysis, PRA's, etc. It will also be used to prepare and analyze LER's and NPRDS data.

SAN DIEGO GAS & ELECTRIC

Very early in their design of the Sundesert Nuclear Power Plant (which was cancelled in 1978), San Diego Gase & Electric Company (SDG&E) initiated an Availability program which could also be termed a Reliability Program. With full SDG&E management support, a specific goal of 90% availability was established, which was to apply to each unit after it had reached maturity (completed 3 years of operation), and a goal for the capacity factor was set at 80%. Availability and capacity factor were defined as follows:

$$\underline{\text{Operating Availability}} = \frac{\text{Available Hours}}{\text{Period Hours}} \times 100\%$$

where

<u>Available Hours</u> = the time in hours during which a unit or major equipment is available for service whether or not it is actually in service

<u>Period Hours</u> = the clock hours in the period under consideration

$$\text{Capacity Factor} = \frac{\text{Total Electric Generation in MWH}}{\text{Period Hours X Maximum Dependable Capacity in MW}} \times 100\%$$

where

Maximum Dependable Capacity = The dependable capacity, winter or summer, whichever is smaller, in MW electric.

Once the goals were established, SDG&E developed a plan to achieve the goals, and an organization with authority and resources to implement the plan. Since they could not find documentation by other utilities or in the literature, SDG&E essentially developed their own plan through the efforts of a consultant. The Sundesert Reliability Engineering Guide (1976) was the resulting document which details the steps to be taken to subdivide the 90% availability goal into pieces small enough to manage. The objective was to identify, in as much detail as possible, those systems and components that cause unit outages, and to quantify those outages, so that priorities could be established for redesign efforts.

The authority and resources were provided through the Sundesert Availability Committee, chaired by SDG&E Nuclear Department Manager, and having one member from the Project Management Office, and one Reliability Engineer from each contractor. SDG&E was further represented by thier consultant, several SDG&E engineers, and the Reliability Engineer action to coordinate the program. Furthermore, SDG&E Management, from the President down, was cognizant of the program, and the responsible Vice President and the Sundesert Project Manager participated in some of the committee meetings.

SDG&E's Reliability Engineering Guide (Appendix C) outlines in detail the flow and structure of the various reliability engineering activities with a brief description of each activity. In addition to the development of the overall reliability program, the program, basically as it developed through 1978, was to convert the availability goal of 90% to an unavailability goal of 10% or 876 hours per year, and then allocate this time to the various critical systems and components. Each contractor was then asked to further subdivide his basic allocations to the system level and then to the component level as information on failure rates, and from reliability analyses, became available.

The basic formal data sources used were

1. Generating Availability Data System (GADS)
2. Licensee Event Reports (LER's)
3. Nuclear Plant Reliability Data system (NPRDS)

The SDG&E Sundesert Reliability Engineering Guide contains analyzed data on the allocation of unavailability for system and subsystems of a nuclear power plant, and preliminary reliability and maintainability allocations of these systems and subsystems. Also, a Reliability Critical Items List (RCIL) was established to focus attention on those components and systems reflecting the highest unavailability. The RCIL was extended to include mean-time-between-failures (MTBF) and mean-time-to-repair (MTTR), and the status of the reliability effort for each item.

TENNESSEE VALLEY AUTHORITY (TVA)

The TVA reliability engineering function for nuclear plants has been assigned to the Nuclear Engineering Branch of the Office of Engineering Design and Construction. The Nuclear Engineering Branch has a Reliability/Availability Section but they do not, as yet, have a formal reliability program. The people who would be used to develop and implement a reliability program are heavily involved in learning how to perform probabilistic risk assessments (PRA's). Thus, they provide technical support to nuclear power operations. As a result, they anticipate that it may take as long as five years before they have an adequate reliability organization and a formal Reliability Program Plan. They feel that they should first become adept at performing PRA's; then, they will be better able to define the elements of a reliability program needed as inputs for a PRA analysis. They have a consultant under contract to: 1) perform PRA's at several of their plants; and 2) to train TVA personnel to perform PRA's so that they have their own in-house capability.

TVA estimates that it takes an average of 15 manyears to perform a nuclear power plant PRA. They feel that a PRA is a decision managing tool that would be useful for "fine tuning" the deterministic design methodology currently in widespread use in the nuclear industry.

PRA methodology will be used quite extensively for each TVA nuclear power plant. Their existing math models are being expanded to provide a full plant model capability. TVA engineers will be trained on how to use the full plant model. Finally, a separate group at TVA will be assigned to keep the full plant model up-to-date, and use it to perform PRA/Safety/Availability analyses.

The PRA currently being performed is concentrating on system interactions, analyzing each system for common mode failures. The focus is on design improvements.

In the past, the safety people at TVA also covered reliability analyses. In performing reliability analyses, they used the exponential distribution of component time-to-failure.

They have developed their own definitions of safety related items, based upon the NRC requirements, with TVA interpretation. They have broken them down into two classes:

1. Primary safety functions
2. Secondary safety functions

Based upon the definitions and classification, each plant prepares a Critical Systems, Structures, and Components List (CSSCL). Any item on the CSSCL requires an LER, if it malfunctions. The first step in preparing an LER is the preparation of Reportable Occurrence Report (ROR) (TVA unique form). Each ROR is reviewed by the Nuclear Engineering Branch to determine if an LER is required.

The safety people also perform the following functions:

1. Do FMEA's and interaction analyses.
2. Write design criteria and review contractor designs.
3. Analyze "backfits" and operating procedures for safety implications.
4. Review operating experience data.

The three basic questions to be answered in the safety analysis of a new design modification are:

1. Does it decrease the margin of safety?
2. Does it increase the probability of an accident?
3. Does it create the possibility of new accidents?

They generally find, and try to fix, most of their problems during plant shutdown for refueling.

They do not review maintenance instructions for safety implications.

TVA is in the process of setting up its own computerized database for component and equipment failure rates. They are using plant specific data as well as generic data from sources such as WASH 1400, NPRDS, MIL-HDBK-217, etc. They merge the plant specific and generic data by means of Bayesian statistical analysis techniques.

Their plan is to ultimately set up a centralized computerized database with remote terminals at each of the plants. The plan is to have a Data Collection and Analysis Center operated by a separate group.

FLORIDA POWER & LIGHT

The Florida Power & Light reliability program is a maintenance management program with an emphasis on a data collection and feedback system called Generation Equipment Management System (GEMS). GEMS translates plant work orders into various computer codes that provide information as to plant unit, specific equipment involved, major and minor equipment codes, action taken, manufacturers, reason/root cause codes, outage hours, power curtailed (in MW), manhours spent in repair, materials and their costs, and contractor cost.

Data are reported in a formal Unit Availability Report. These reports are submitted for all plants. In a quarterly report, all systems or items which led to an outage (took a plant off-line) during the quarter are identified, together with the cost of the occurrence and the probability of loss. From this report, overtime, mean times between failures, mean times to repair, unavailabilities, and costs are generated; these then lead to a critical items list--items which require attention. Each item is given to a specialist within FP&L to fix. Thus, the data from the Unit Availability Reports provide an essential starting point for improvement efforts.

In June, 1982, they hope to initiate a probabilisitc analysis of operationally critical systems; they will supply their data to a fault tree analysis program.

The nuclear and fossil programs are on the same level, use the same systems, and have the same philosophy. There is no reliability person or program especially for nuclear, just for the total system.

GEMS was established in 1971 and tied to work orders in 1972; so FP&L has 10 years of good data with which to work. Over the 10 years, FP&L has shuffled the program around and solved most of the problems. About three years ago, it was decided to initiate no new programs, but to make use of what they already had.

FP&L then does not have a formal reliability program, per se, just the informal program described above. They tried to get a reliability person on staff to develop a future reliability group, but the position was cut out of the budget, so they're sticking with the informal program for now. The emphasis of the program is on operations and maintenance, especially a well-planned, scheduled maintenance program. They have no formal Reliability Centered Maintenance (RCM) program, but a maintenance management system based on performance losses and boroscope inspections. Time is a factor in maintenance scheduling, but it is tempered by other factors.

A report entitled "Power Resources Department Reliability Program", February 9, 1979, contains a list of all the things that FP&L does in the area of reliability, not a description of a formal program. They do issue "Reliability Reports on ...." to guide them in decision making on various issues, i.e., which system to buy, whether to backfit, etc. For these reports, they use mainly their own formal reliability data. They sometimes use NERC or NPRDS data.

They do not use reliability formally in procurement, but they do use historical reliability data to select vendors.

DUKE POWER COMPANY

Duke Power Company does not, at this time, have a formal reliability program. They are just starting one; it won't really be in place until the end of the summer. The program will be geared to increasing plant availability. They are currently in the data collection phase of the program to determine which way to go.

Essentially, they are collecting failure and repair time data on the components and systems in each operating plant. These data will be analyzed to determine the "cost drivers" which are contributing to reduced plant availability. They will then prioritize the "cost drivers" and develop and document an Availability Improvement Program tailored to each plant.

Their primary concern is to improve the reliability, availability, and maintainability of existing plants. As they view it, there are three phases of a reliability program:

1. Visual inspection for obvious design and manufacturing defects.
2. Detailed analysis of design, operating, and maintenance procedures to uncover areas for improvement.
3. Formal reliability program.

They have been concentrating their efforts on the first two phases, and are about to enter the third phase.

Their approach is to identify problems and feed them back to the design engineers for correction. Their feedback mechanism is a Feedback Guide which is provided to the designers. Essentially, it is a "lessons learned" document which contains detailed descriptions of problems encountered in operation and the corrective action recommended. Hopefully, by using the Feedback Guide, which is a sort of corporate memory, the designers will avoid making the same mistakes in the future.

The definition of plant availability that they use is:

$$\text{Availability} = \frac{\text{Source Hours + Reserve Shutdown Hours}}{\text{Total Period Hours}}$$

They do perform some R&D reliability studies. They are currently developing a reliability prediction technique for steam generator tube failures based upon a Weibull distribution of time-to-failure. Reliability numbers, e.g., MTBF, are beginning to be used in some of their studies, but they do not specify reliability numbers in their equipment specifications.

They are currently developing what will ultimately become a computerized component data collection and analysis system. It will be utilized for performing PRA's. It will use:

1. Plant specific data from the Work Request System.
2. NPRDS and LER data.
3. Data form in-house component testing.

MIDDLE SOUTH UTILITY SYSTEM (MSUS)

MSUS has developed an Availability Improvement Program (AIP) which has the goal of minimizing plant life-cycle costs. The key elements of the AIP are:

1. Capability to conduct system/unit/component level quantitative availability analyses.

2. Capability to conduct quantitative unit life-cycle analysis.

3. Computerized data system to support 1 and 2.

The AIP is divided into a short term program and a long term program. The short term program is aimed at the use of available data to determine major problems and availability improvements that might be made by solving those problems. The long term program includes the following elements:

1. Unit Model Baselines (UMB's)

   o evaluate equivalent availability (EA)

- o mathematically model the hardware
- o rank problem areas by impact on EA
- o provide basis for Unit Availability Investigations (UAI's)
- o provide input to Cost Benefit Analysis (CBA's).

2. Unit Availability Investigations (UAI's)

   - o focus in specific unavailability problem areas
   - o determine root causes of unavailability problems
   - o determine feasible solutions to unavailability problems
   - o evaluate risks associated with solution implementation
   - o provide inputs to CBA's.

3. Cost Benefit Analyses (CBA's)

   - o accept inputs from UMB's and UAI's
   - o evaluate economic benefit of proposed solution on a life-cycle cost basis.

4. Availability Data System (ADS)

   - o provide AIP analysts with the analytical tools necessary to conduct availability improvement studies.

5. Training of MSUS engineers in AIP methodologies

The long term program was scheduled to be implemented over a five year period, beginning 1979. As of the middle of 1981, the following had been accomplished:

1. 12 UMB's completed

2. 8 UAI's completed or in progress

3. CBA in progress

4. 21 AIP specialists on board, 5 at MSNA, the rest spread almost equally across the four operating utilities.

The computerized Availability Data System consists of the following subsystems:

1. Generating Availability Data Reporting System (GADRS)

   - o Data acquisition system (just went on line)
   - o NERC/GADS data report generation
   - o NPRDS data report generation

2. Basic Analytical Cycle System (BACS) (Computerized UMB)

3. Cost Benefit Analysis (CBA)

4. Unit Availability Investigation (UAI)

5. Advanced Analytical Cycle System (AACS)

   - GO (developed by EPRI, based upon point estimates of component failure and repair rates)

   - RAM (uses probability distributions and confidence intervals for failure and repair rates--if available)

The complete system will not be on line until 1984; as of this time, they are in the process of implementing the data acquisition phase of the Generating Availability Data Reporting System.

CONSOLIDATED EDISON

Consolidated Edison has an advanced reliability and maintainability (R&M) program, transferring DoD/NASA R&M technology to Consolidated Edison, which has been developed over the past 10 years. The work encompasses most of the elements of a good R&M Program.

Reliability engineering has provided the greatest return to Consolidated Edison in the following areas:

Failure Analysis

The root cause of repetitive failures has been determined in several areas. Deeper and more scientific analysis succeeded where past techniques had not. These skills are being brought in-house.

Failure Mode and Effect Analysis

Rigorous, step-by-step analysis of the ways that systems can fail and the effect of each failure (with probabilities) has resulted in design changes and improved operating procedures.

Design Review

Reliability engineering design review has resulted in the establishment of specific contractual requirements for reliability, availability and maintainability. It has also permitted introduction of human factors considerations resulting in the reduction of both operating and maintenance manpower. The chances of "operator error" have also been reduced. These benefits are the result of review from a viewpoint different from pure system function.

Data Management

Data systems have been set up that, while collecting reliability and engineering data, are used to provide information to operating departments. The information has made possible significant productivity increases and manpower savings. The data capture and retrieval methods introduced by reliability engineering have brought computer utilization and benefit directly to operating personnel.

Spare Parts Analysis

Using failure and repair statistics as a basis for selection of spares has reduced cost and increased equipment availability. Vendor recommendations often tend toward expensive and/or high-profit items. This new approach uses failure and repair statistics to optimize the choice of spare parts.

Failure Prediction

Quantitative predictions of probability and frequency of failures and determinations of the useful life of equipment have resulted in more cost-effective purchases. They have also been used to good effect in capital equipment and manpower budgeting.

Life-Cycle Cost Analysis

Analysis of the total cost of equipment, including purchase price and cost of installation, operation, maintenance and removal for replacement has shown some surprising results. Lower life-cycle cost for equipment has resulted.

They have 9 in-house people (5 direct, 4 indirect) to support the reliability program and a contractual budget (~$250K) which enables them to tap any one of up to 100 outside specialists to do R&M studies.

Consolidated Edison purchases some equipment against quantitative reliability specifications. See Appendix E for an example of an equipment reliability specification used by Consolidated Edison. The reliability specification is based upon MIL-STD-785, "Reliability Program for Systems & Equipment, Development and Production," which is used in DoD procurements. It has been tailored to Consolidated Edison requirements.

One approach to equipment reliability specifications is to use Reliability Improvement Warranties (with incentives and penalties) based upon a life-cycle cost analysis. Consolidated Edison uses a 5-year warranty for specified availability, leaving the contractor to establish the MTBF/MTTR trade-offs to achieve the specified availability.

Consolidated Edison intends to have at least 6 equipment reliability specifications within the next 6 months. Future plans call for the development of a generic reliability specification from which individual equipment reliability specifications can be tailored.

For reliability predictions, Consolidated Edison uses the exponential distribution to time-to-failure because of its simplicity, relative accuracy, and mathematical tractability.

Thus, the basic thrust of their program is to find the "cost drivers," and fix them to increase plant availability.

Consolidated Edison does have an internal Qualified Vendor's List from which they make their equipment buys based upon previous, good experience.

Consolidated Edison has in operation a rather sophisticated reliability and maintainability data collection and analysis system for their plants. It is not an in-house system; they buy it from a time-sharing organization.

To improve the adequacy and accuracy of their database, Consolidated Edison is developing a computerized Power Plant Maintenance Improvement System (PPMIS).

The application of the data collection and analysis system is to increase plant availability, perform trend analysis studies, and do PRA's.

NORTHEAST UTILITIES (EU)

NEU has a Reliability Engineering Group whose primary goal is to increase plant availability. The program emphasis is on determining those components which are the major contributions to plant unavailability and recommending corrective action.

The main elements of their program are:

1. Thermodynamic analyses of plants to find and locate components with cause power degradation.
2. Vibration signature analysis to detect impending failures.
3. In-service inspections to detect impending failures.
4. Root cause analysis studies to discover and correct the basic failure modes and mechanism.

Although it is not yet in place, NEU is planning to develop and apply a Reliability Centered Maintenance (RCM) concept for its plants in order to optimize the maintenance process for maximum plant availability. They are currently conducting an economic study using reliability data to develop an optimum procedure for buying spare motors. The study will be completed in 6 months. NEU's policy on plant availability improvements is that the payback must occur within one year; this is because of the problems that some utilities are having in obtaining investment capital.

NEU has a Qualified Vendor's List which is under the purview of the Quality Assurance (QA) Department. QA surveys prospective vendors who are not on the list to insure that they will provide products of the required quality.

NEU does some equipment qualification testing in their Electrical Engineering Group; they are primarily concerned with the aging of electrical insulation.

NEU does not use reliability specifications to buy equipment. However, they support the concept, and are planning to use it in the future.

NEU has participated, and will continue to participate, in the Interim Reliability Evaluation Program (IREP) in which a reliability evaluation is being conducted on two of their plants. They anticipate that the two studies will require a total of 10 manyears.

NEU has conducted a PRA on Milestone 1. The Fault Tree Analysis (FTA) computer codes are currently on the NEU computer. NEU is developing an in-house capability to do PRA's; they are also training their designers to make PRA's part of the design process.

NEU is in the process of developing a computerized plant data collection and analysis system to develop MTBF and MTTR data in order to increase plant availability.

In addition to plant-specific data, they plan on using data from:

1. NERC/GADS
2. NPRDS
3. NRC Graybook
4. LER's
5. IEEE STD 500

They are currently collecting the data manually and putting it into the database. The nuclear plant database is in pretty good shape (from a computerized standpoint); the fossil plant data is still, mainly, manual.

CAROLINA POWER & LIGHT COMPANY

The Carolina Power & Light reliability efforts at corporate level are concentrated in one individual. Others at corporate level are also performing reliability type functions, but primarily directed toward improving safety and efficiency. It has always been a "goal" to set up a formal Corporate Reliability Program, but it has never been done. Primarily, the reliability efforts have been directed toward design and development of individual reliability programs at each of individual CP&L plants.

The Nuclear Safety Group at corporate level has assumed some reliability functions such as data collection, reviewing LER's from NRC and using NOTEPAD. These efforts have primarily been directed at meeting NRC requirements, and have absorbed the limited resources available. The Safety Group also does trend analysis to determine root cause of failures and assess if failure is due to normal wearout, random failure or overstressed conditions. They then attempt to determine if a better part is available.

CP&L also collects data and performs analyses of plant operating conditions such as downtime, forced outages, maintenance records, and quality assurance programs, in order to improve plant efficiency and maintenance scheduling. However, these data are not analyszed as part of a formal Reliability Program. CP&L's Corporate Material Control Division uses the data for inventory control, sparing, some quality control, and provides direction to plant reliability engineers to work with manufacturers on specific problem areas. A Preventive Maintenance Management System has been established to look at all outage data to adjust and prep an preventive maintenance. Also, CP&L works with Combustion Engineering to evaluate root cause of failures and make recommendations (CP&L makes decision if they can, or should, implement the recommendations); Babcock and Wilcox looks at unit outage reports and also makes recommendations.

CP&L has a formalized Reliability Program at each of its operating plants. This program is fully supported and directed from corporate mangement, and has been inoperation since mid-1980. The reliability activities are carried out within each plant. A corporate level reliability engineer develops and administers the overall program, but, primarily, he assures that the individual programs are active. This involves assessing the reliability reports, providing feedback and communication between individual programs, reviewing data and literature applicable to programs, and providing training and assistance as required.

The Reliability Program at the individual plants is divided into three activities: miscellaneous administrative, preventive, and corrective. The program is administered and directed by a reliability engineer at each plant, who is supported by the plant engineering and operating staff as needed.

The preventive activities are directed toward proper operation of the plant; the basic premise being that plant and equipment should be operated in full accordance with the manufacturer's operating and maintenance procedures. This includes analysis of operating data to assure operation within manufacturer's specifications, calibration and verification of control and measurement equipment, and developing O&M procedures to fit specific conditions. They have found these measures to work well in that most problems are resolved within the first three years after startup of a new plant.

The heart of the CP&L active Reliability Program is their corrective activities which is essentially a formalized, systematic program to take care of problems which would improve availability. Each year the plant manager provides an estimate of the equivalent availability goal, based on scheduled outages, and a list of specific projects to be undertaken to achieve the goal. He then justifies his efforts in meeting the goal through reliability improvement projects. This does not necessarily mean that availability will go up during a given year, since unscheduled operating outages, etc., can make availability better or worse than the goal.

The projects to be considered for the reliability program corrective activities are normally the more significant problems within the plant. If the job can be handled by a "work ticket," it is not considered a reliability problem. New projects can be added to the program at any time. The work tickets are reviewed and, if repeated jobs appear, they might be a source for a reliability project.

The Reliability Program corrective activities are formalized in that the plant manager is required to hold a monthly meeting and provide a written report on progress on projects underway, and new projects being undertaken. The person responsible for the problem equipment, or area, is assigned responsibility for its soulution. He can assign appropriate persons to investigate the problem, and research and effect a solution. He may obtain help from line managers, reliability or engineering personnel, consultants, or other expertise from within CP&L. The plant manager sets priorities for projects and is responsible for evaluating both short term and long term results of the corrective action. The plant reliabiltiy engineer oversees the projects for the plant manager. Monthly written reports are provided to corporate level management.

CP&L also has a Plant Efficiency Program which is similar to their Reliability Program, only directed toward unit heat rate and other efficiency goals.

APPENDIX C

RELIABILITY ENGINEERING GUIDE
SAN DIEGO GAS AND ELECTRIC COMPANY

During the design of the Sundesert Nuclear Plant, Units 1 & 2, which were canceled in 1978, the San Diego Gas and Electric Company developed a Reliability Engineering Guide. The guide presents the Management and Reliability Engineering Activities which were to be implemented to achieve high availability, capacity factor and system reliability.

The Guide is reproduced in this Appendix but the Appendices to the Guide are not included.

D. W. LATHAM

Sundesert Nuclear Plant
UNITS 1 & 2

RELIABILITY ENGINEERING GUIDE

October 1976

Project Manager
San Diego Gas & Electric Company

CONTENTS

APPENDIXES

INTRODUCTION

It is the goal of the San Diego Gas & Electric Company (SDG&E) to design, construct, and operate the Sundesert Nuclear Plant to meet the highest practical levels of availability, capacity factor, and system reliability.* Target availability and capacity factors at plant maturation are set at 90 and 80 percent, respectively.

To achieve these goals, a systematic reliability engineering effort will be implemented. The purpose of this guide is to stimulate and identify a specific plan of action covering the design phase of the Sundesert Nuclear Plant (SNP). Guides for other phases of the plant life cycle will follow. The implementation of the guide is to proceed as an integral part of the project design effort. While the guide covers only design, to be successful its elements must be sustained throughout fabrication, construction, and operation of the SNP. The guide applies to all SNP systems required for the production of electrical power and for the prevention or mitigation of accidents. Implementation of the guide shall be in accordance with other applicable project control documents; industry standards; and Government regulations, guides, and standards.

The effectiveness of application of the guide is strongly dependent on organization, planning, data considerations, and training. These elements are discussed in the context of management, the most fundamental quality of any plan.

MANAGEMENT

SDG&E shall have overall responsibility for managing implementation of the Reliability Engineering Program (Figure 1). The SNP Reliability Engineer shall have direct access to his counterparts in the contracting organizations. The SNP Project Manager shall be responsible and accountable for the success of the program in achieving the availability goals. Day-to-day execution of the program will be the responsibility of the SNP Reliability Engineer.

The reliability engineering function:

- Formulates plant availability goals and the related reliability and maintainability goals necessary to achieving availability goals.

*Definitions of terms in Appendix A.

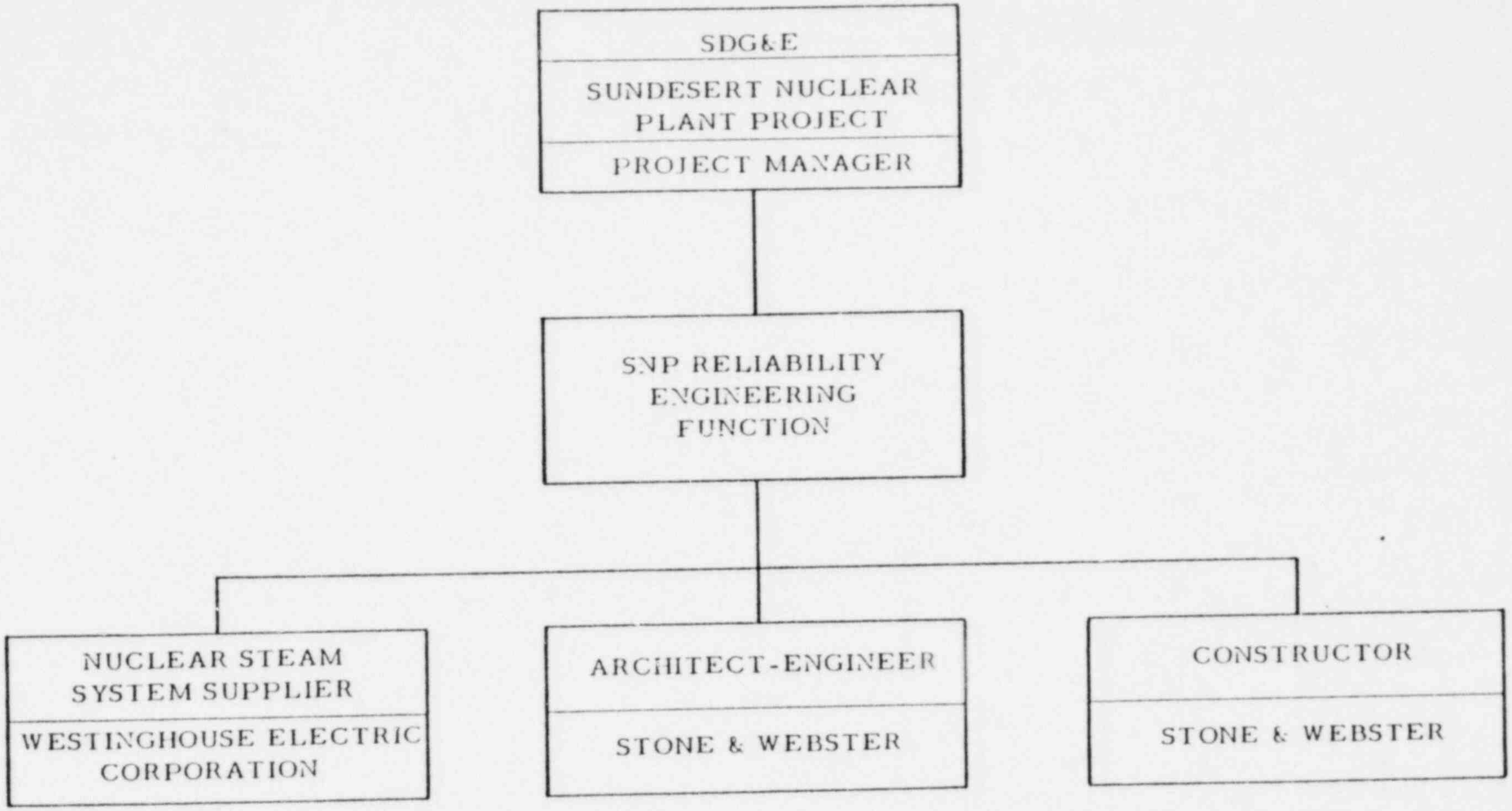

FIGURE 1

ORGANIZATION STRUCTURE
RELIABILITY ENGINEERING - SUNDESERT NUCLEAR PLANT

- Establishes a systematic plan for implementing reliability goals through all phases of the design project.

- Evaluates design characteristics, system and equipment concepts, and operating and maintenance plans to assure that goals are achievable.

- Provides tangible outputs such as trade-off studies; reliability/maintainability inputs to specifications; reliability evaluations for safety reports; maintainability design guidelines; and reliability requirements for specific research, development, and testing.

- Directs and evaluates reliability training and indoctrination programs for SDG&E and contractors.

- Resolves problems identified by contractors in meeting reliability requirements.

- Reviews and approves/concurs adequacy of reliability requirements:

  - Procurement contracts and specifications.

  - Contractor work plans/packages.

  - Contractor specifications, drawings, analyses, and other documentation.

  - Contractor data sources.

  - Test programs and operating and maintenance procedures.

- Participates in design reviews.

Each SNP contractor shall be responsible for preparation and implementation of an SDG&E-approved reliability program plan for use on SNP systems for which he has lead design responsibility. The guide and the resulting program procedures, reviews, analyses, etc., shall be in accordance with the contract between SDG&E and the contractor. A primary requirement of each contractor program shall be that the contractor's reliability function have direct access to the contractor top management and the SDG&E/SNP reliability function for timely resolution of special problems.

RELIABILITY ENGINEERING ACTIVITY

The reliability engineering activity for the Sundesert Nuclear Plant is outlined in Figure 2. The outline is not intended to be exhaustive. The approach is to be as brief as possible only highlighting the main elements of the reliability engineering effort being applied to Sundesert. To give focus to the engineering and design activities, such other matters as management and personnel, training, and procedures are only briefly mentioned. The activity description which follows is to be interpreted in the context of its application. For example, the application depends on the system involved and the particular stage of the design effort. For some systems certain of the activities require an extensive amount of analysis while for others only minimum investigation is necessary.

Figure 2 has been developed to highlight what are considered to be the most critical elements of reliability engineering. These highlights correspond approximately to the top row of the activity network. Feedback and iteration loops are shown to indicate continuous adjustments in inputs and outputs as the design progresses and as new information is developed. No attempt has been made to quantify the feedback loops; nor does the absence of a particular feedback necessarily mean that it does not exist. As a start, and based on limited design data, a preliminary allocation has been made for Sundesert together with a first cut at a reliability critical items list (RCIL). The allocations have been made both in terms of unavailability (Appendix B) and availability (Appendix C). Where data permitted, consideration was given to subsystems (e.g., turbine blades) of first line systems (e.g., turbine). A preliminary reliability critical items list is presented as Appendix D.

Figure 2 presents the flow and structure of the various reliability engineering activities. A brief description of each activity is given below. The numbers match those of Figure 2.

1. Set Availability Goal for Plant

   It is the goal of SDG&E to design, construct, and operate the SNP to meet the highest practical levels of availability, capacity factor, and system reliability. Target availability and capacity factors at plant maturation are set at 90 and 80 percent, respectively.

2. Perform Systems Analysis

   To perform reliability engineering, it is necessary to define the system to be engineered. The total system of concern includes

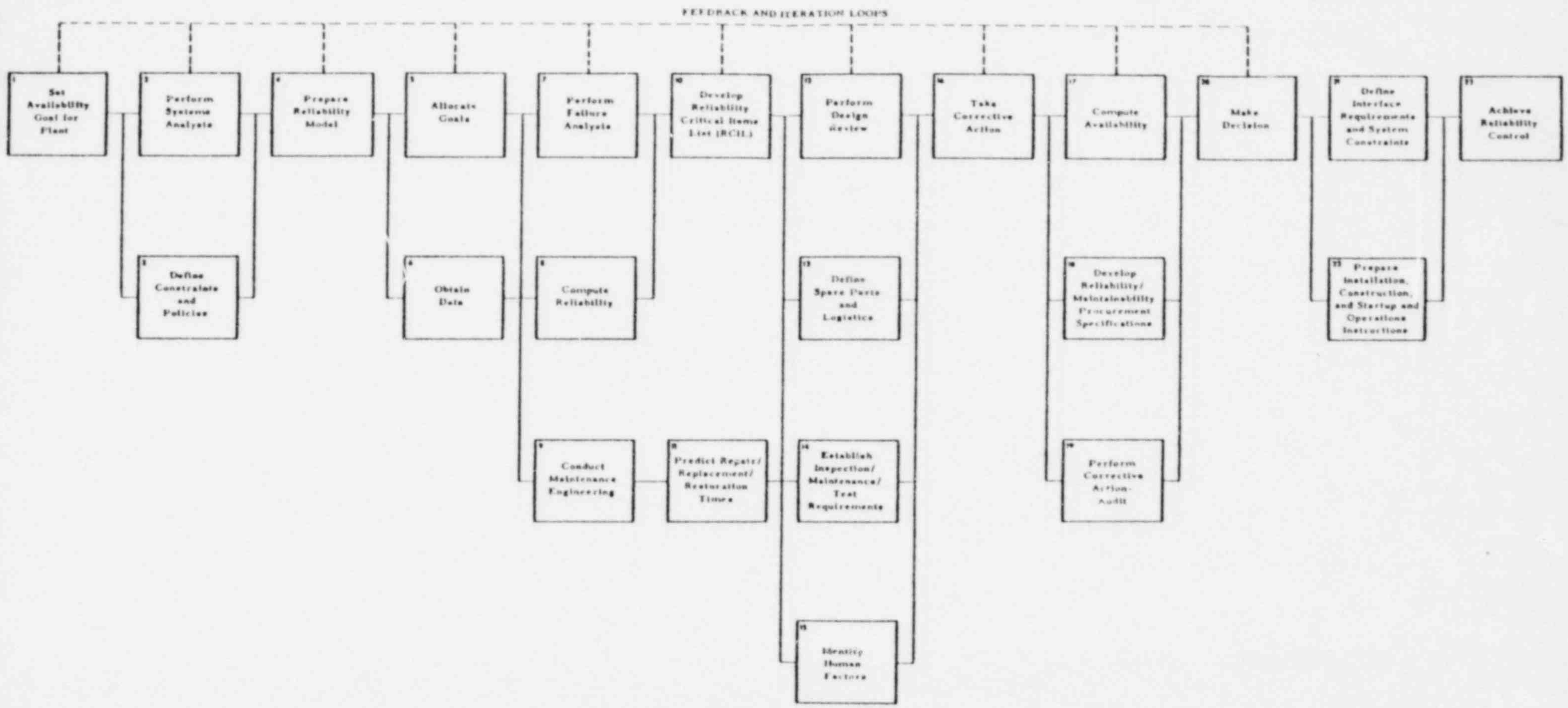

FIGURE 2

RELIABILITY ENGINEERING
ACTIVITY NETWORK
SUNDESERT NUCLEAR PLANT

the nuclear island, the energy conversion system, the containment, the plant site, and supporting software. Be it the total system or a subsystem such as the feedwater system, it is necessary to determine the functional relationships of components and subassemblies. The systems affecting reliability are identified and functional and general physical descriptions are developed. Information developed will be system equipment configurations, including redundant and standby functional relationships and functional dependencies within and between systems. This activity provides a concise description of principal design criteria, operating characteristics, and reliability implications. Of course, this is an activity continually updated as the design progresses.

3. Define Constraints and Policies

The impacts of safety, construction, and operating constraints on the system are delineated. Operating, maintenance, and testing policies are described and made visible to the design and reliability engineering effort.

4. Prepare Reliability Model

The purpose of the model is to define the reliability relationship of components and systems. The models may vary from simple block diagrams to complex logic diagrams involving fault trees or event trees. The model need not be complex. A simpler model, consisting of a simple diagram and a few equations, is adequate for many purposes. Selected techniques are highlighted in Appendix E. The main objective at this point is to begin a mechanism of ferreting out reliability intensive components and systems. The model will provide a basis for design trade-off studies, design selection, and components/parts selection.

5. Allocate Goals

The overall plant goal will be allocated (apportioned) to the lowest system/component level justified by the stage of the design. While there are numerous methods of allocation (see Appendix F), early in the design it is often adequate to allocate according to complexity and predicted failure rates and repair times. The main purpose of the allocation is to translate reliability requirements into understandable design goals at a level where they can have the most impact on system performance. Allocation is the inverse process to a reliability prediction. In fact, it is an iteration between the two that converges to the most meaningful allocation. As the design effort continues, the allocation becomes increasingly refined.

6. Obtain Data

There are numerous data sources available (Appendix G). Using these and other sources of failure rate and repair data, appropriate data are to be selected for each significant failure mode in systems/components and personnel. If environmental factors are available, they too should be used. These data are used as predictive input and should be the best available. When data are lacking, estimated failure and repair rates are used and are so identified.

7. Perform Failure Analysis

Failure analysis deals with the system fine structure and, in particular, is performed to isolate and identify weaknesses in the design. Numerous qualitative and quantitative methods are available (Appendix E). Failure analysis will progress from system to component level and from failure mode analysis and fault hazard analysis to common mode and fault tree analysis. Failure mode and effect analysis developed in the military programs is the best known. FMEA is a qualitative technique which involves the identification and tabulation of the ways (or modes) in which a part/component can fail. The final objective of failure analysis regardless of method is the determination of ways to eliminate or reduce the probability of incidence of critical failure modes to improve the design. It is the key activity for exposing corrective actions.

8. Compute Reliability

The combination of the reliability model (Step 4) and the failure analysis (Step 7) provides a meaningful basis for predicting reliability. The approach is usually to develop equations for series-parallel components representative of the system (exactly, or, if the system is complex, an approximation is usually made). Because of the high level of complexity of nuclear systems, the reliability computation is often waived in favor of computing availability. This is because steady-state availability is simpler to compute and in fact contains reliability. If the system happens to be simple, and many nuclear plant subsystems are, then it is useful to make the reliability calculation. In any event, this is the step for refining the reliability parameters, most notably, failure rates and operating conditions.

9. Conduct Maintenance Engineering

Maintainability engineering will be performed to define spares allocation and to predict repair, replacement, and restoration times. The necessary operation and maintenance actions to keep equipment or systems operating will be defined. Analysis will be performed to evaluate the degree of achievement of the maintainability design goals, including logistics and personnel interaction. Particular attention is to be given to equipment access and simplicity of required operations.

10. Develop Reliability Critical Items List (RCIL)

A reliability critical items list will be generated and maintained identifying and ranking those systems/equipments/components contributing most to uncertainties in meeting reliability goals. Inputs to the RCIL shall be based on failure analysis and reliability computations. A key factor in making the list is equipment whose failure could cause an unsafe condition or could cause loss or impairment of availability for power production either directly or as a result of long repair or replacement times. Therefore, criticality is based on the role of the equipment in the system (obtained from the system model) and its failure and repair characteristics (obtained from data and failure analysis). Early in design and for the hierarchy systems, it is possible to perform a criticality ranking just based on a judgment of the uncertainties of the system and its overall importance to reliability. For example, a preliminary RCIL for Sundesert first line systems was developed and is presented in Appendix D.

11. Predict Repair/Replacement/Restoration Times

Repair times of equipment will be predicted to the lowest level in the system consistent with the progress of the design. The planned repair method will be determined and estimates of equipment restoration times made.

12. Perform Design Review

A design review board will periodically evaluate critical designs and considerations to provide additional assurance that the equipment and systems are capable of achieving the performance and reliability requirements. A primary resource for performing the review will be the RCIL. Thus, design reviews will be a forum for reviewing checklists, design changes, test results, and impact of reliability analysis. The "design" in a design review is the set of drawings, layouts, specifications, performance predictions, analyses, and system design descriptions that describe what is to be manufactured, ordered, installed, and operated. The design review of items on the RCIL shall be supported by design, failure, and maintainability studies. These studies shall verify that the design is acceptable from a reliability/ maintainability standpoint or shall identify problem areas which must be resolved to obtain acceptable design and possible solutions. Any problems revealed that would prevent the achievement of allocated availability goals will result in a "HOLD."

Members of the design review team shall be selected or approved by SDG&E who will also establish meeting schedules and agendas.

13. Define Spare Parts and Logistics

Maintenance and reliability considerations will be the basis for providing adequate spares to permit transfer of operable equipment from supply to use and failed equipment from use to storage. Operating and maintenance logistics are to be based on achieving plant availability goals.

14. Establish Inspection/Maintenance/Test Requirements

Based on the maintainability and reliability analysis, inspection intervals and test requirements will be established. These requirements will be compared with the goal allocations and the assumptions made in the availability computation. Maintenance requirements will be established to assure system operability and reliability. These requirements will address procedures, laydown space, and resources. The inspection and test requirements are to be defined in terms of scope, duration, and frequency.

15. Identify Human Factors

Studies will be made of the interaction requirements between people and equipment/systems to assure achievement of performance goals. Emphasis will be on operations and maintenance. In particular, during design, people interaction considerations will be given to all conceivable modes of the equipment/system including normal operations, outages (planned and forced), and accidents. Human factors will become an integral part of achieving reliability control.

16. Take Corrective Action

Based on information presented for design review and the recommendations of the design review board, a corrective action plan will be developed. A log will be maintained of all known reliability (and other) problems with recommended solutions and names of persons responsible for carrying out the action. The implementation of corrective actions shall be traceable.

Options to be considered in corrective actions include:

- Examination of data sources to validate need for corrective action.
- Testing to verify that need for corrective action is valid.
- Selection of more reliable equipment and systems surveillance techniques.
- Use of redundancy.
- Use of shorter test inspection and maintenance intervals (for reliability improvement).
- Redesign for shorter repair times.
- Elimination of operator/maintenance induced failures.
- Addition of physical or procedural safeguards.
- Reallocation of goals to eliminate need for corrective action.

Appendix H is a list of those systems and items which past experience indicates as frequent contributors to unavailability.

17. Compute Availability

The failure and reliability analysis together with predicted repair times enable the system/plant availability to be computed. Availability involves both reliability and maintainability. The results from the availability calculation provide a measure of the ability to meet the system goals. Substantial information now exists for making decisions about system performance.

18. Develop Reliability and Maintainability Procurement Specifications

Engineering and equipment specifications are among the most important tools for achieving visibility and control of reliability and maintainability. The intent is that the specification should be written to assure that a procured equipment meets its reliability and maintainability goals. Reliability and maintainability specifications can vary widely depending on the role of the equipment, its complexity, and industry experience in the application of reliability engineering. It is intended that, wherever possible, quantitative requirements be specified. Such requirements include failure rates, repair times, inspection intervals, environmental factors, and spares. In some cases, demonstration or testing will be required.

19. Perform Corrective Action-Audit

An audit function will be implemented which initiates and monitors corrective action and reliability improvement.

20. Make Decision

Based on the availability computation, decisions can be made to accept or reject designs, to change the reliability goals, or to make refinements in the reliability engineering effort. Naturally, decision points will occur at several different stages of the design as well as along several different points in the reliability process.

21. Define Interface Requirements and System Constraints

Reliability control is very dependent on a clear understanding of system constraints and interfaces. Too often is heard the expression "our system was caused to fail by somebody else's." System interfaces and constraints are to be defined in terms of such considerations as instrumentation and control, allowable thermal and mechanical loads, process conditions including water chemistry, and compatibility of person-machine interactions.

22. Prepare Installation, Construction, and Startup and Operations Instructions

All is lost unless the reliability and maintainability state-of-mind that is cultivated during design is sustained throughout construction and operation. An important element of this process is, during design, to prepare instructions for later phases of the plant life cycle which are truly rooted in reliability and performance considerations.

23. Achieve Reliability Control

The result of this systematic reliability engineering effort is a continuing achievement of plant reliability control. The visibility of reliability activities enhances project communication and management.

APPENDIX D

LRM
CRBRP RELIABILITY
PROGRAM PLAN (DRAFT)

The Clinch River Breeder Reactor Project developed a detailed description of the essential elements of their Reliability Program Plan. Included are the principal LRM activities, organizational responsibilities, management structure and management tools and procedures to be used to assure high reliability.

A draft copy is reproduced in this Appendix.

LRM
CRBRP RELIABILITY
PROGRAM PLAN

March 20, 1981
REV. 0

Prepared by:
D. W. Giles
D. W. Giles
LRM LIcensing

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FIGURES AND TABLES

CRBRP Reliability Program Plan

1.0 INTRODUCTION

The purpose of this document is to identify the principal LRM activities, organization responsibilities, management structure, and the management procedures and tools used to assure that the program objective of enhancing the CRBRP design through integration of the reliability discipline and techniques is achieved. The reliability program requirements are specified in OPDD-10 and the program and all its associated activities are described in Appendix "C" of the PSAR. This document will detail the essential elements of the program briefly but primarily will describe the LRM's implementation of the CRBRP Reliability Program.

2.0 PROGRAM DESCRIPTION

Each of the potential sources for release of radiological species has been evaluated. The results of this evaluation determined where reliability program resources should be applied in order to maximize the health and safety benefit to the public. Program activities are focused on preventing the loss of coolable geometry in the reactor core. The rationale supporting this conclusion is summarized in Appendix "C" of the PSAR.

The success of two different missions is necessary and sufficient to maintain coolable core geometry. One is the reactor shutdown mission which encompasses the plant protection systems capability to detect and process critical plant parameter anomalies into shutdown commands and the primary and secondary control rod systems capability to respond to those commands by insertion of sufficient negative reactivity worth to successfully shut the reactor down. The other is the shutdown heat removal mission which encompasses the shutdown heat removal systems capability to bring the reactor from initiation of shutdown to thermal equilibrium at standby. The reliability program focus is on systematic failure analysis and assessment of components, subsystems and systems* whose failure to function could prevent or degrade mission success.

A Reliability Related Components List (RRCL) is baselined in an ICD to identify those plant equipments that can contribute to the success or failure of the safety mission. It organizes the safety critical equipment of systems into logical groups, subsystems, or components for analysis and documentation.

There are two distinguishable analytical efforts to the overall program:

1) The qualitative analysis effort which is structured by the RRCL to align with the component and system design responsibilities of engineering, and,

2) The quantitative analysis effort which models the success logic of the overall Reactor Shutdown System (RSS) and the failure logic of the overall Shutdown Heat Removal System (SHRS) to allow system failure probability predictions.

* SDD level

The qualitative effort is an integral part of the design development process. The emphasis is on the identification, analysis, and closeout of the failure modes of the RRCL components and systems. Areas of concern or uncertainty are identified by the analyses for resolution. These analyses are scheduled to support the design development and review process. The findings and conclusions from the analyses are summarized and presented at the component and systems design review. The final results of the qualitative assessment program are documented in Reliability Design Support Documents (RDSDs).

The quantitative effort is concerned with modeling the failure logic of the RSS and the SHRS. Failure rates are assigned to the component failure modes. When the failure logic is math modeled, computerized studies are conducted to evaluate the reliability of the overall RSS and SHRS systems and their sensitivity to changes in operating assumptions, systems configurations, failure probability assignments, etc. On a comparative basis the model studies provide valuable insights into the system interactions and dependencies. These studies aid in an understanding of the design and operational characteristics of the subsystems and systems, and they support the design decision making process. The Reactor Shutdown System and Shutdown Heat Removal System quantitative reliability assessments are conducted and documented independently from the qualitative RDSD documentation program. Insights are exchanged between the qualitative and quantitative assessment programs through the Controlled Information Data Transmittal System.

There are other program elements which influence or are influenced by the reliability analysis program. The most important of these are the design performance, verification and qualification test programs that are being conducted on RSS and SHRS equipment. Early reliability program analysis efforts are influential in the structure of these test programs. The validity of the reliability assessments are predicated on successful test program results. If failures occur, corrective actions are developed and verified by engineering and the impact of these failures on the numerical predictions are evaluated by the reliability organization. Other CRBRP project elements are also important to the reliability assessments. They include the quality control programs, qualification program, inservice inspection and testing, operations, maintenance, and configuration control programs which effect the plant safety-related equipment.

3.0 PRINCIPAL ACTIVITIES

3.1 Introduction

Design and performance requirements, industry standards, regulatory criteria and guidance are imposed by OPDD-10 and expanded into detailed requirements in SDD's and equipment specifications. The reliability program is designed and structured as illustrated in figure 3.0.1 to proceed in parallel with the design effort. Established reliability methods and analytical techniques are employed that have been used in DOD, NASA, nuclear industry, and non-nuclear industry programs. They include failure mode and effects analysis, common cause failure analysis and logic modeling and numerical assessment. These analyses are conducted to identify component and system failure modes, to analyze their effects, and to assess the adequacy of component design features to minimize potential failures. These analyses also evaluate system design features (e.g. redundancy) to mitigate component failure consequences on the system

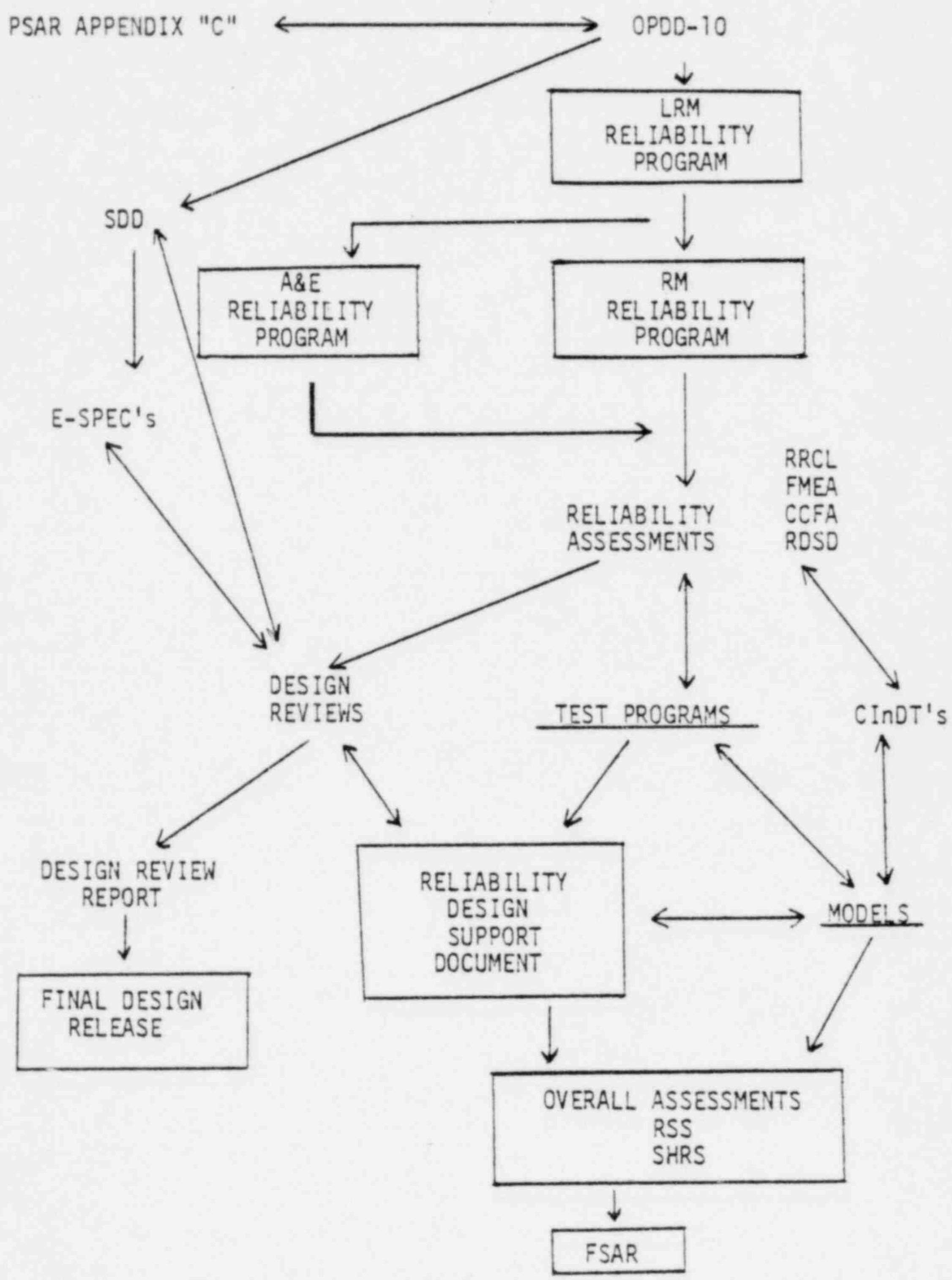

Figure 3.0-1 PROGRAM ACTIVITY STRUCTURE

function. This analytical effort is integrated into the design development process to maximize early awareness of areas that require corrective actions. Program findings that impact the design are integrated into the program throughout the process. The resolution of reliability program findings along with other design support disciplines are reviewed and approved or otherwise dispositioned at the equipment and system design reviews.

Logic diagrams for the overall RSS and SHRS are constructed, quantified and math modeled. This allows numerical assessment of the RSS and SHRS probability of failure and supports identification of the systems dominant contributors to failure and the sensitivity of the system to modeling assumptions and changes. This modeling effort and the qualitative assessment proceed in parallel and the insights gained from each effort are exchanged.

3.2 Qualitative Analysis

The Safety Related Reliability Program requirements are specified in paragraph 7.2 of OPDD-10. Systems and selected subsystems or components which have the potential to degrade or prevent the safety functions of reactor shutdown or shutdown heat removal will be qualitatively analyzed as an integral part of the design development process and the analyses will be documented as a necessary part of the design documentation.

3.2.1 Reliability Related Components List (RRCL)

The Reliability Related Components List (RRCL) is the baseline list of the systems and lower level equipment that require a reliability analysis in support of the equipment at design reviews. Figure 3.2-1 lists the reliability program for the RSS and SHRS functional systems specified in OPDD-10. The systems specified in OPDD-10 are analyzed and the list is indexed to a lower subsystem, component, or component group level appropriate for separate analysis. The reliability related components listing is baselined to the appropriate analysis level and maintained in Interface Control Document ICD CL54012.

3.2.2 Failure Mode and Effects Analysis

Failure Mode and Effects Analysis (FMEA) is the reliability analysis methodology that provides a disciplined approach in the search for random failure modes and identifies their functional effects on a component, subsystem, system or overall systems.

The FMEA format identifies the part, or assembly, and its function. It lists the failure modes that could prevent or degrade the safety-related function. For each failure mode identified causes are listed and their effects are assessed. The rationale for acceptance of the design, test, or procedural features that control the potential failure mode are developed or additional actions required to control the failure mode are developed. The additional actions required to resolve significant concerns or uncertainties are tracked to resolution. FMEAs are performed at the component or logical component group level, and at the system (SDD) level. At a

OVERALL PLANT DESIGN DESCRIPTION

(OPDD-10)

REACTOR SHUTDOWN SYSTEMS (RSS)

- SYSTEM 31 REACTOR SYSTEM
- SYSTEM 32 REACTOR ENCLOSURE SYSTEM
- SYSTEM 90 PLANT CONTROL SYSTEM
- SYSTEM 92 REACTOR AND VESSEL INSTRUMENTATION SYSTEM
- SYSTEM 95 FLUX MONITORING SYSTEM
- SYSTEM 99 PLANT PROTECTION SYSTEM

SHUTDOWN HEAT REMOVAL SYSTEMS (SHRS)

- SYSTEM 31 REACTOR SYSTEM
- SYSTEM 32 REACTOR ENCLOSURE SYSTEM
- SYSTEM 51A PRIMARY HEAT TRANSPORT SYSTEM
- SYSTEM 51B INTERMEDIATE HEAT TRANSPORT SYSTEM
- SYSTEM 52 STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM
- SYSTEM 53 STEAM GENERATOR SYSTEM
- SYSTEM 56 REACTOR HEAT TRANSPORT INSTRUMENTATION & CONTROL SYSTEM
- SYSTEM * 68 PIPING AND EQUIPMENT ELECTRICAL HEATING & CONTROL SYSTEM
- SYSTEM 81 AUXILIARY LIQUID METAL SYSTEM
- SYSTEM * 82 INERT GAS RECEIVING & PROCESSING SYSTEM
- SYSTEM 90 PLANT CONTROL SYSTEM
- SYSTEM 99 PLANT PROTECTION SYSTEM

* INTERFACE ANALYSIS ONLY TO EVALUATE INFLUENCE ON SHRS.

NOTE: ANALYSIS OF THESE PRIMARY SYSTEMS IDENTIFY IMPORTANT INTERFACING SECONDARY SYSTEMS THAT REQUIRE ANALYSIS. THESE ARE CONTROLLED BY INTERFACE DATA TRANSMITTALS.

FIGURE 3.2-1 SAFETY-RELATED SYSTEMS

minimum a FMEA is required at each RRCL component and system level design review. A summary of the reliability assessment and its findings is presented to the review team. The necessary actions for open concerns and uncertainties are identified and entered into the Project Centralized Action Correspondence Control System and tracked to a final resolution.

These analyses identify the need for changes to the design, test, analysis, or procedures for the component or system or provide additional assurance that the design is adequate. In their final form the System Reliability Design Support Documents provide a record of the projects qualitative reliability program and its findings.

3.2.3 Common Cause Failure Analysis

Common Cause Failure Analyses (CCFA) are conducted to evaluate functionally redundant trains for system susceptability to failure as a result of single events. The CRBRP shutdown and shutdown heat removal systems have been designed to have redundancy for critical functions that is independent and where appropriate, diverse. The purpose of CCFA is to ensure that no single events have been overlooked that could fail the systems safety related functions.

Checklists of historical common failure causes are tailored to the unique characteristics of the redundant systems operating environment and function. A systematic search is conducted for common cause failure susceptability in each redundant train. The random failure modes identified by FMEA are examined for potential occurrence due to identified commonalities. Significant concerns or uncertainties are identified for project review and action as required.

CCFA is performed at the component level when appropriate and at the system (SDD) level. Common dependencies are entered into the overall system models (e.g. electrical power sources, environmental conditioning, etc.). External phenomena (i.e., flood, fire, earthquake, tornado, etc.).are design basis events for safety-related equipment which are also included in CCFA. The qualitative CCFA addresses the adequacy of the design features and, when quantifiable and considered significant, are treated in the system level numerical assessment. The CCFA results provide additional assurance that safety-critical functions considered to be redundant are in fact redundant. Potential events or system conditions that could nullify redundancies are identified for corrective action. The significant findings of CCFA's are integrated into the design/development process for CRBRP.

3.2.4 Interfacing Systems

Interface dependencies are identified and are analyzed in component and system level FMEAs and for possible single point failures that violate redundancies in CCFAs. The primary interface list in the SDDs serve as a checklist of the interface requirements needed from support systems, and provide the link for continuity of coverage from system to system.

The RSS and SHRS modeling programs evaluate the systems across interfaces. The information developed in the qualitative analyses support and verify the accuracy of the quantitative model from interface to interface. The project Controlled Information Data Transmittal (CInDT) system is used to insure the cognizant design engineer's review and concurrence with the SHRS model for operating assumptions and reasonableness of the failure probabilities. The modeling programs for the RSS and SHRS include consideration of system interactions and dependencies on Balance of Plant systems (BOP). The SHRS model is used to develop a list of Balance of Plant (BOP) systems that are supportive to the shutdown heat removal system. The list is ranked by importance to success of the SHRS mission. It provides guidance in focusing the BOP reliability efforts on the BOP systems that support the SHRS.

3.2.5 Reliability Design Support Document

The Reliability Design Support Document (RDSD) records the complete qualitative reliability assessment of equipment identified on the RRCL.

It provides a description of the equipment, its primary functions and special design requirements, and describes its role in or its interface with the system and identifies potential consequences resulting from failures. Each RDSD also defines the analytical methods, and input data. It incorporates the FMEA and CCFA and summarizes their results. Special design analyses, test program results, quality programs, operations, and other supporting program activities are discussed. Conclusions are presented and significant concerns or uncertainties, if identified and unresolved, are highlighted with recommendations and actions for their resolution. They provide a project record of the qualitative safety-related reliability programs, scope, methods, findings, conclusions and the final disposition of reliability-related concerns. RDSDs, supplied as part of design review package, are prepared for the system at the SDD level. The level at which analyses are conducted and documented is selected to support the component and system design review level. Selected lower level groups of equipment are analyzed and reliability assessments are provided for the lower level design reviews. The remaining system equipments are analyzed in detail in their systems' FMEA and are summarized in their system level RDSD. CCFA's are conducted at the system level and when appropriate at the component level. Lower level assessments are included or summarized in their system level RDSD. The purpose of the RDSD is to support the design development process and to record the qualitative reliability program and its results. RDSD's are included as part of the final system design review data package.

3.3 Quantitative Analysis

The overall reliability of the RSS and the SHRS will be assessed quantitatively.

Complex models have been constructed that graphically represent the success/failure logic for the RSS and the SHRS. When quantified, the models provide a tool for identifying the systems significant contributors to failure probability. They permit studies that aid in understanding the systems sensitivities. Sensitivity studies are conducted to explore input uncertainties (e.g. failure rate assignments, operating assumptions, etc.) or may be used to evaluate the value of design or operational options.

Source data for model construction includes, but is not limited to Project design documentation, the qualitative reliability analyses (FMEA, CCFA, RDSD, etc.) from the RMs and the B&R reliability programs, and the test programs. Adjustment of historical data for use in the models is based on engineering judgement considering experience with similar mechanisms.

The functional balance of the design from a reliability viewpoint is assessed. The graphics of the models are an aid in qualitative evaluation of the system for subtle interactions and dependencies. Base case predictive values of system failure probability are calculated.

The models are constructed at the RSS and SHRS mission levels. The potential failure contributions of all primary, secondary, and auxiliary systems important to the RSS or SHRS missions are considered in the models. The contributions to system failure probability at any level can be determined.

Sensitivity studies with the SHRS model will be conducted to quantify the contribution to SHRS failure probability of each B&R system to the individual functional interface level. The functional interface list will be ranked by importance allowing minor contributors to be eliminated from the B&R Reliability Related Critical Interface List (RRCIL).

The value of the modeling and quantitative assessment program is in its use as a design tool. It is a valuable aid in understanding a complex system and the importance of its component parts to the overall systems mission. Its purpose is to assist the plant designers in the development of a balanced design of safety-related equipment for reliability.

3.3.1 RSS Quantitative Assessment

The quantitative reliability assessment of the Reactor Shutdown System conservatively predicts that the RSS will operate orders of magnitude more reliably than the SHRS. In order to approach the objective of a balanced design for CRBRP reliability quantitative program resources are being concentrated on the SHRS. However, this assessment shall be reviewed relative to the findings from the test program and the on-going qualitative reliability analysis program. As the design development analysis and test programs are completed, the assessment shall be reviewed by engineering and reliability to determine if an update is required. Other program results that influence reliability (qualitative programs, waivers/deviations, key system reviews, etc.) shall be considered in these reviews.

3.3.2 SHRS Quantitative Assessment

Modeling and quantitative assessment of the Shutdown Heat Removal System is a continuing process throughout the design development and test programs. The SHRS dependencies, interactions and dearth of a relatable empirical data base demands careful evaluations to ensure that the significant contributors, their operating and failure characteristics, and failure data uncertainties are properly addressed. It is an iterative process of establishing a base case prediction, identifying the dominant contributors to unreliability, examining the contributors through sensitivity studies, refining the modeling assumptions to account for additional plant capability, or making design/procedural modifications and re-establishing the base case. Design reviews of the model are held and special meetings of design and model specialists are convened to ensure realistic treatment of the significant contributors to SHRS failure probability and to identify corrective actions as appropriate.

3.4 Supporting Programs

The results from the design/performance verification and qualification test programs for critical systems are factored into the qualitative and quantitative reliability assessment programs. Successful design performance development test programs support the failure data judgements that were developed from historical experience for similar equipment in similar operating environments. The Reliability Program related equipment test programs are monitored to assure that reliability related requirements are satisfied through the failure reporting analysis and corrective action programs, with the aid of the unusual-occurrence reporting system, and by the day-to-day interactions of reliability with the responsible engineering organizations. The results of test programs and other project activities pertinent to the reliability of the RRCL equipments are discussed in the development and final documentation of the reliability program results in system RDSDs. Consideration shall include related information internal and external to the project such as quality assurance, change control, FFTF experience, base technology tests, foreign breeder experience, LWR experience, and NRC interactions. The final Reliability Program and its findings will be summarized in the Final Safety Analysis Report.

4.0 RELIABILITY PROGRAM STRUCTURE

Project Office policy, budget allocation, and schedular constraints are interpreted, the programs' objectives identified, the methods developed, and the activities of the participants are defined and coordinated to ensure implementation of an integrated CRBRP Reliability Program.

The LRM has fiscal and technical management responsibility for the Safety-Related Reliability Program. The scope of the LRM's program includes the NSSS systems that are the design responsibility of the LRM. This includes identification of the reliability-related interfacing NSSS and BOP systems for which Burns and Roe (B&R) has design and reliability analysis responsibility. These B&R supportive system interfaces are identified initially by qualitative review of the reliability SHRS and RSS modeling work. The list of identified system interfaces is issued by the LRM reliability program for guidance in support of the B&R reliability program. The LRM has the responsibility to review the B&R reliability analysis and integrate the results into the overall reliability program.

The equipment in the LRM systems that require qualitative analyses are identified by ICD CL54012 which baselines the Reliability Related Components List (RRCL). Table 4.0-1 lists the type of analyses that are performed for each item on the list and identifies the responsible RM.

W-RM has overall responsibility for the modeling and numerical assessment program for the Reactor Shutdown System. The GE-RM has overall responsibility for the modeling and numerical assessment program for the Shutdown Heat Removal System. These responsibilities include obtaining sufficient understanding of the equipment to integrate the functional and interactive failure characteristics of the complete system; and researching the failure rate histories of like or similar equipment in order to assign realistic failure probabilities to the failure modes of interest. All participants in the LRM's qualitative analysis program and B&R are responsible for providing support to the modeling organizations in the form of consultation on request and by comprehensive review of the modeling of their equipment to ensure logical portrayal of operating characteristics and reasonableness of the failure rate assignments. The Project Controlled Information Data Transmittal System is used to provide discipline and a record of significant information reviews and exchanges between contractors. In this manner, the important interactions and failure probability effects of interfaces between the SHRS equipment and their supporting systems are factored into the quantitative assessment.

TABLE 4.0-1

RELIABILITY RELATED COMPONENTS LIST (RRCL)

GENERAL ELECTRIC

	FMEA	CCFA	RDSD
RSS			
Secondary Control Rod System System 31 (FSWDR 31WS139)	X	X	X
Secondary Control Rod Drive Mechanism Driveline	X		
Secondary Control Assembly	X		
SHRS			
Intermediate Heat Transport Loop System 51B (FSWDR 51G1M07)	X	X	X
Pumps (Intermediate and Primary Loops)	X	X	
Steam Gen. Aux. Heat Removal System System 52 (FSWDR 52G1M11)	X	X	X
Protected Air-Cooled Condenser	X		
AFWS	X		
AFW Pumps	X		
Steam Gen. Loop System 53 (FSWDR 53G1M11)	X	X	X
SG Modules (W-LRM Responsibility)	X	X	
Rupture Discs	X		
H_2O/Dump Valves	X		
Leak Detection Modules	X	X	
Relief and Isolation Valves	X		
Steam Drum	X		
Recirculation Pumps	X	X	
Reactor Heat Transport Instrumentation System 56 (FSWDR 56G1M10)			X
Sodium Pump Drive	X	X	
I&C - 51, 52, & 53	X	X	
Piping & Eqpt. Elec. Heating & Control System 68 (FSWDR 68G1M09) Interfaces with SHRS/RSS			X

TABLE 4.0-1

WESTINGHOUSE - ARD

	* RA (Incl. FMEA)	RDSD (Incl. FMEA/CCFA)
RSS		
Reactor System System 31 (FSWDR 31 W5139)		X
Upper Internals		X
Upper Internals Structure	X	
Upper Internals Structure Jacking Mechanism	X	
Lower Internals		X
Core Support Structure	X	
Lower Inlet Module	X	
Core Former Structure	X	
Horizontal Baffle	X	
Fixed Radial Shield	X	
Core System		X
Fuel Assemblies	X	
Blanket Assemblies	X	
Removable Radial Shield Assembly	X	
Core Restraint System	X	
Primary Control Rod System		X
Primary Control Assembly	X	
PCRDM and PCRD	X	
Shield & Seismic Support	X	
PCRS Maintenance Equipment	X	
Reactor Enclosure System System 32 (FSWDR 32W[illegible]03)		X

* Reliability Assessment (RA), Preliminary to RDSD.

TABLE 4.0-1

WESTINGHOUSE - ARD (Cont.)

	RA (Incl. FMEA)	RDSD (Incl. FMEA/CCFA)
Reactor Enclosure - Head System		X
Reactor Vessel Closure Head	X	
Riser Assembly	X	
Plug Drives and Controls	X	
Head Heating and Cooling System	X	
Bull Gears and Bearings	X	
Reactor Enclosure - Vessel System		X
Reactor Vessel and Supports	X	
Reactor Enclosure - Head Access Area		X
Flux Monitoring System Mechanical Components	X	
Flux Monitoring System System 95 (FSWDR 95WA305)		
Primary Linear Power Range Channels	X	
Secondary Linear Power Range Channels	X	
Plant Control System System 90 (FSWDR 90WA750)		
PPS Control and Display Components		X
Primary CRDM Power and Control Equipment		X
Scram Breakers	X	
Primary CRDM Controller	X	
Primary CRDM Motor Generator Sets	X	
Reactor and Vessel Instrumentation System System 92 (FSWDR 92WA932)		X
Plant Protection System System 99 (FSWDR 99WA778)		X

TABLE 4.0-1

WESTINGHOUSE - ARD (Cont.)

	RA (Incl. FMEA	RDSD (Incl. FMEA/CCFA)
Primary Instrument Channels	X	
Secondary Instrument Channels	X	
Primary Logic	X	
Secondary Logic	X	
HTS Pump Trip Circuits	X	
SGAHRS Initiation Logic	X	
Monitor	X	
SHRS		
Reactor System System 31 (FSWDR 31WS139)		X
Lower Internals		X
Core Support Structure	X	
Horizontal Baffle	X	
Reactor Enclosure System System 32 (FSWDR 32WK503)		X
Reactor Enclosure - Head System		X
Reactor Vessel	X	
Reactor Vessel Support	X	
Primary Heat Transport System System 51A (FSWDR 51WF200)		X
Guard Vessels		X
Reactor Vessel Guard Vessel (System 32)	X	
Reactor Guard Vessel Extension (System 32)	X	
IHX Guard Vessel (System 51A)	X	

TABLE 4.0-1

WESTINGHOUSE-ARD (Cont.)

	RA (Incl. FMEA)	RDSD (Incl. FMEA/CCFA)
Primary Sodium Pump Guard Vessel (System 51A)	X	
Piping and Check Valves Hangers and Snubbers	X	
Intermediate Heat Exchanger		X

ATOMICS INTERNATIONAL

SHRS	RDSD (INCL. FMEA)	RDSD (INCL. FMEA/CCFA)
Auxiliary Liquid Metal System System 81 (FSWDR 81A4080)		
Direct Heat Removal Service (DHRS)		X
Air Blast Heat Exchangers	X	
EM Pumps	X	
Remotely Operated Valves	X	
Pressure Vessels and Piping	X	
Overflow Heat Exchanger	X	
DHRS System Instrumentation and Control System		X
Inert Gas Receiving and Processing System System 82 (FSWDR 82A0293) Interfaces with SHRS/RSS		X
Liquid Metal to Gas Leak Detection System System 66 (FSWDR 66A1145) Interfaces with PHTS		X

5.0 RELIABILITY PROGRAM IMPLEMENTATION

This section defines essential tasks, organizational responsibilities and management controls employed by the LRM to assure implementation of the reliability portion of the total Reliability Assurance Program for the Clinch River Breeder Reactor Plant (CRBRP) Project. This section applies to Project participants herein identified and defines interface responsibilities within the LRM.

5.1 Reliability Program

Authorized reliability program elements for CRBRP are shown in Table 4.0-1 of section 4. Section 5 will describe the organizational relationships, resources, and procedures employed by the LRM to assure that the program participants fulfill their responsibilities for the program elements.

5.2 Organization

Figure 5.2-1 shows the LRM organizational relationships pertinent to the implementation, coordination, and management control of the technical and fiscal aspects of the Reliability Program as authorized in applicable CRBRP Project documentation.

5.2.1 Resources

In addition to appropriate facilities and management policy and practices sufficient to establish and implement the authorized reliability program,personnel resources authorized and allocated to the program include a reliability program staff engineer who reports to the Licensing Manager. The RM Program Staff Managers organizations which include LRM cognizant plant systems engineers and the Systems Engineering and Integration staffs are available resources as required. LRM Quality Assurance, Program Control, and Procurement provide support to the reliability program as needed.

5.2.2 Communication

Formal reliability program communication external to the LRM is prepared, reviewed, approved, and controlled within the LRM in accord with approved procedure LX-OR-2. Receipt, transmittal, and distribution of program correspondence is controlled by LX-OR-3, and correspondence files and logs are maintained in accord with LX-OR-4. Internal LRM reliability program day to day informal coordination is communicated either by internal memorandum or personal contact.

5.3 Responsibility

5.3.1 Responsibility of Participants

Fiscal management responsibility for the reliability program at the RMs is maintained by the LRM's three RM Program Managers.

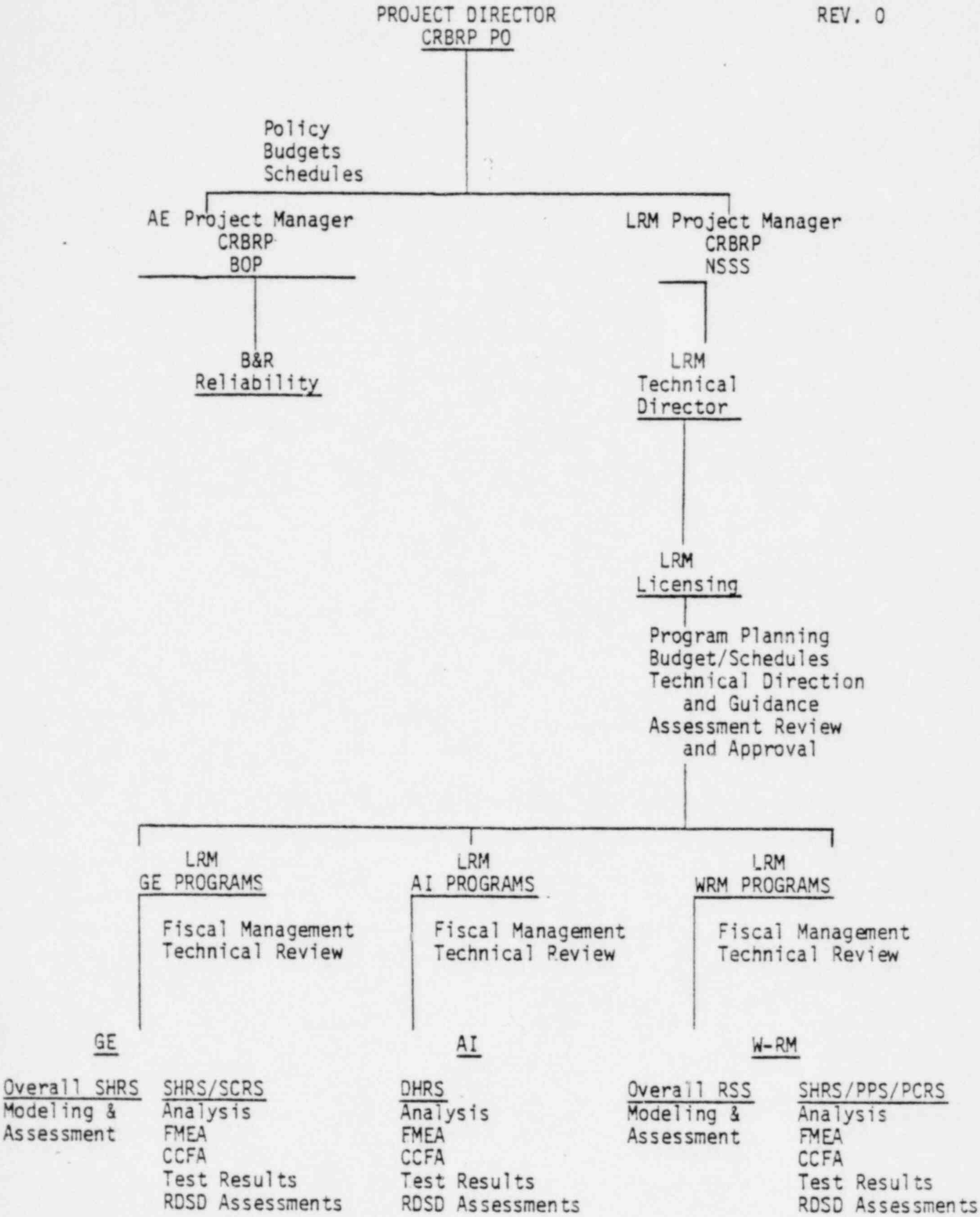

Figure 5.2-1 Program Organizational Responsibilities

Technical management responsibility is with the manager of Licensing. The reliability staff reports to the Licensing Manager and is responsible for defining and maintaining the program's scope, technical content, methodology and schedules within CRBRP budgetary and design schedule constraints. The reliability staff perform overall program planning, integration, coordination, and surveillance of the RM programs. The execution of the reliability program is the responsibility of RM's in accordance with their design responsibilities for plant systems. Figure 5.2.1 shows the organizational responsibilities for the program.

5.3.2 Interface Relationships

There are internal LRM interface relationships and external LRM to RM, A E, and Project Office interface relationships in the implementation and execution of the reliability program.

Internal LRM interfaces exist between the reliability staff and all other LRM organizations. Vertical interfaces are from the reliability staff upward through the Licensing Manager, the Project Technical Director to the Project Manager. An additional vertical interface exists between the reliability programs and the RM Program Managers. The RM Program Managers maintain responsibility for overall RM fiscal control including the reliability program. They also have review and concurrence responsibility for technical content and accuracy of reliability program products. This is ensured by the LX-OR-2 requirement that direction and guidance is issued to an RM over the signature of the RM program manager; a practice that provides formal coordination of the programs direction with the appropriate cognizant engineers for plant systems. Horizontal interface relationships exist between the reliability staff and all other LRM organizations on an as required basis. The reliability program requires informal day to day coordination activity with systems integration and engineering, plant systems cognizant engineers, quality assurance, program control, document control and others.

External LRM reliability program interfaces exist with the other program participants and the Project Office. Formal correspondence is over the signature of the RM program managers or the CRBRP Project Manager. Informal day to day activity is coordinated and communicated by telephone, data-fax, pouch mail, or personal contact as required to optimize program effectiveness. The Project Controlled Information Data Transmittal system as described by LX-OR-10 is used to exchange technical data between the performing participants.

5.3.3 Design Implementation

The primary objective of the reliability program is achievement of a optimum balanced design for safety-related reliable operation of CRBRP. This is achieved through early identification of significant concerns and implementation of needed design and, procedural corrective actions throughout the design development process.

Actions required to correct identified deficiencies are developed through coordinated action by reliability and design. Schedules for implementation are developed and action commitments are entered into the project Centralized Action Correspondence Control System in accord with LX-OR-5. This procedure ensures a final resolution of the uncertainties and significant concerns identified through the reliability program effort.

5.4 Planning

This document establishes the overall LRM Reliability Program Plan. The plan along with the appendix contains the policy, practices, and LRM management activities required to implement the program. It describes the programs objective, scope, contents, methods, and the requirements and guidelines necessary for execution by the program participants. The individual internal plans, practices, and procedures of the program participants are developed and provided as information to the LRM. Since the participants (RM's) actually conduct the analyses and document the assessments more detailed step by step procedures are developed for performing the program activities to an established format. To the extent practical and appropriate the LRM provides guidance for achieving uniformity of the various participants products.

5.5 Document Control

The program participants have internal document control for documents related to the analytical requirements of the program that shall include: preparation; identification; review and approval; issue; distribution, and storage; and revision. Final system and Reliability Design Support Documents prepared by the RM's are subject to review and approval by the LRM. Lower level analyses and assessments are subject to internal RM approval and are forwarded to the LRM for information on the progress of the program and for identification of generic programmatic or technical problems that require corrective action. Final system level RDSD's are circulated for review and comment within the LRM before LRM approval of the documents. Procedure LX-OR-8 provides control and a record of the review. The LRM reliability organization maintains a schedule, index, and file of the program reliability assessment documents.

5.6 Review and Audit

5.6.1 Reliability Program Management Reviews

The LRM reliability organization participates in management reviews of the participants reliability programs. These may be requested by the RM, the LRM or the Project Office. They may be called to address specific problems or results, or to provide program status to management.

5.6.2 Audits

The LRM conducts audits of the RM's Reliability Programs in conjunction with and as an extension to the existing Quality Assurance audit system as defined in LX-OR-46. LRM Quality Assurance conducts the audit and the LRM reliability staff either participates as an audit team member or as an observer.

5.7 Reporting

Reliability Program status meetings are held between the LRM and Project Office staff on approximately a monthly basis to report the status, progress and significant results of the program.

5.7.1 Reports

Final System RDSDs summarize and report the qualitative reliability for each safety related system. Quantitative reliability assessment reports are provided for the overall RSS and SHRS. The LRM will provide an overall program summary report that documents the important findings from the qualitative and quantitative reliability assessments.

APPENDIX

RELIABILITY PROGRAM

REQUIREMENTS AND GUIDELINES

APPENDIX A

RELIABILITY PROGRAM REQUIREMENTS AND GUIDELINES

INTRODUCTION

This Appendix is an extraction and organization of reliability program Requirements and Guidelines from letters of direction by the LRM to the Program Participants. The letters are indexed for reference at the end of the Appendix. The direction has been classified as requirements or guidelines based on "what" is required to satisfy program objectives (requirements) and acceptable methods for "how" to do it (guidelines). With W-LRM approval, methods the RMs consider more appropriate than the guidelines may be acceptable. The requirements and guidelines are organized into categories depending on the program activities they apply to. They are indexed by category item number (first digit) and reference letter list identifier (second digit).

REQUIREMENTS AND GUIDELINES

GENERAL

Requirements

1.2 Qualitative design support assessments shall be performed to identify, evaluate, and disposition RRCL component failure modes.

2.2 RRCL component and system assessments shall be performed on a schedule to support the design review process.

3.3 The significant concerns and uncertainties summarized in assessments shall be extracted and the actions required for their resolution and final disposition shall be scheduled and transmitted to the LRM for entry as an REL item in the "Centralized Action Correspondence Control System".

4.3 If there are procedural constraints identified during the assessments that would prevent unsafe operation of the system, they shall be documented in the system RDSD and extracted for tracking and final resolution with the procedure writers.

5.1 Each RRCL item shall be controlled by an E-Specification (or equivalent) which delineates as a minimum complete physical, environmental, and performance requirements; reliability and quality assurance requirements including inspections and tests for qualifications, acceptance, and lot sampling where required; explicit requirements to be satisfied in accepting parts and for packaging, handling and traceability.

6.1 To the extent available at the time of scheduled review, reliability design support documentation shall be incorporated into the formal design review packages and reviewed at formal design reviews.

7.1 Information copies of all reliability assessments (interim and final) shall be forwarded to the W-LRM and Project Office Reliability Staffs as they are completed.

8.1 Final RDSDs shall have W-LRM concurrence prior to release.

9.4 It is emphasized that the Safety-Related Reliability Program as specified in the RMs internal procedures is to be an integral part of the design development process. The baseline for the program is the RRCL. The FMEA, CCFA, and RDSD for RRCL components shall be scheduled to support the design/development process (i.e. design reviews, design release, fabrication release, etc.).

10.7 The critical interfaces identified by component and system level FMEA shall be reviewed against the SDD primary interface list to ensure adequate specification of the interface requirement in the SDD and ICDs.

11.4 Each analysis preliminary to the final RDSD shall be summarized to highlight the significant findings mutually agreed to by the engineering and reliability analysts.

a) The finding may be that the equipment design is considered adequate at this stage of the design process to prevent or mitigate all identified failure modes.

or:

b) Significant concerns or uncertainties with the equipment's design or operation that require additional actions for resolution were identified,

and:

c) The agreed upon actions associated with these concerns or uncertainties shall be identified, scheduled, and reported to the W-LRM for entry into the CACCS as a reliability (REL) item for tracking to resolution.

Guidelines

1.2 Qualitative design support assessments should be the results of a combined effort by design engineering and reliability engineering.

2.2 Qualitative insights gained from RRCL item design support assessments compared to modeling assumptions, failure rate assignments, and failure mode coverage should be transmitted to the modeling organization.

3.4 Each assessment (preliminary, update, final) associated with the RRCL should be scheduled relative to key design activities and entered into PMS-IV for program visibility.

4.3 Assumptions should be avoided when possible. If they cannot be avoided, they should be verified in the same manner as significant uncertainties (CACCS system entries).

5.3 The FMEA and CCFA should document the rationale for adequacy of failure mode control by discussion of design features, analysis results, and test program results.

6.3 Any equipment specifically excluded from an assessment should be identified and a statement as to where it will be treated shall be provided.

RELIABILITY RELATED COMPONENTS LIST (RRCL)

Requirements

1.1 The RMs, AI, GE and ARD shall be responsible for development of the RRCL for those items that are under their design responsibility.

2.1 All components or functional elements in the RSS, PPS, and SHRS shall be evaluated to identify those that perform a vital function with respect to safe shutdown and/or shutdown heat removal.

3.1 Items shall not be excluded from the RRCL based on probability discrimination.

4.1 Revisions to the RRCL ICD shall be transmitted with justification to the W-LRM.

5.1 Components with Category 2 or 3 failure modes (degrades the RSS or SHRS function, or prevents the RSS or SHRS function) shall appear on the RRCL.

6.4 If at any time, equipment omissions are identified that should be on the RRCL a change to the RRCL ICD shall be proposed.

Guidelines

1.1 The listing should be organized and indexed by functional groups that are logical for further analysis and documentation (FMEA, CCFA, RDSD) under the major subsystem headings of RSS and SHRS.

2.1 The initial vital functional element listing should be subjected to preliminary FMEA analysis to identify component failure modes specific to safe shutdown and/or shutdown heat removal.

3.1 Each failure mode identified by FMEA should be categorized as (1) has no effect on RSS or SHRS function, (2) degrades the function of RSS or SHRS, or (3) prevents the function of RSS or SHRS.

4.1 Components that only have Category 1 failure modes should not appear on the RRCL, but a record of the analytical rationale that supports their categorization shall be retained by the RM and should be available on request.

5.4 Periodically, the RRCL should be reviewed for continued validity.

FAILURE MODES AND EFFECTS ANALYSIS (FMEA)

Requirements

1.2 FMEAs (identification, evaluation, and disposition) shall be conducted on RRCL items.

2.2 For those RRCL item failure modes that are not fully dispositioned by existing analysis results, test results, or design features, a plan of action for resolution shall be identified as part of the assessment.

3.2 All identified failure modes shall be dispositioned by documenting the design features that control or prevent the failure or its effects. The justification shall include results of analyses and tests and any special checkout or inspection procedures or system redundancies that could prevent or mitigate the effects of the failure.

4.4 The component level FMEA shall:

a) Identify all failure modes of components including those resulting from malfunction of interfacing equipment inputs.

b) Identify the system (SDD) level FMEA which will address the interfacing system effects and evaluate the criticality of impacts and design ramifications.

5.4 The system level analysis shall evaluate the criticality of the interfacing system impacts and disposition or recommend resolution of component failure modes in question based on criticality to the system function.

6.4 The assessment (a FMEA and its summary as a minimum) shall be a constraint on conduct of design reviews.

7.5 Each primary interface, listed in the RRCL components System Design Description (SDD) shall be reviewed. Failure of the RRCL components caused by interface effects shall be documented in the components FMEA.

Guidelines

1.3 Vendor supplied RRCL item equipment requires FMEA analysis in support of RRCL item assessments. These may be supplied by the vendor or conducted in-house. The vendor should be required to provide failure experience on like or similar equipment when it is available.

2.1 Each Category 2 and 3 failure mode should be evaluated to identify the design features that control the failure mode.

3.1 The uncertainty associated with the design features that control failures should be evaluated by review of the design development program analyses and test results.

4.1 Significant residual uncertainties should be resolved by additional analyses or test or be brought to the attention of the appropriate design review board for discussion and disposition.

5.2 The system modeling and quantitative assessment effort should be supported during conduct of FMEAs by review of failure rate data used in the math model. A qualitative evaluation of its validity should be made. Recommended changes with supporting rationale should be forwarded to the modeling RM.

COMMON CAUSE FAILURE ANALYSIS (CCFA)

Requirements

1.2 Common cause failure analysis (CCFA) identification evaluation and disposition shall be conducted in a systematic fashion on RRCL items.

2.2 RRCL items with a susceptibility to credible common cause events shall be highlighted in the assessment and a plan of action developed for disposition.

3.3 The random failure modes identified by FMEAs of equipment performing redundant safety functions shall be assessed for susceptibility to potential common cause factors.

Guidelines

1.1 A functional path (system to component) CCFA should be performed for critical functions of the RSS, SHRS and PPS to identify failure potentials in the system due to multiple component failures initiated by a common event.

2.1 An analysis of the system and/or component design features that tend to preclude each common cause failure should be conducted and an evaluation of their adequacy made.

3.1 Those common cause failure potentials that are judged to have a significant residual uncertainty should be resolved by additional analyses or test or be brought to the attention of the appropriate design review boards for discussion and disposition.

4.2 A listing of historical common failure causes should be compiled. The likelihood of the identified causes originating in the components system or its interfacing systems should be qualitatively assessed.

5.2. The capability of a component or system to withstand a common cause initiator should be assessed by investigating failure modes for design features and their operating margins which will accommodate the event and prevent failure.

6.2 The failure rates used in the systems math model should be reviewed to determine if a common cause failure rate factor should be added. Those that are identified should be brought to the attention of the modeling RM.

7.3 Common Cause Failure is defined as a concurrent failure of either identical or non-identical components that perform redundant functions, due to a single common causative factor. Concurrent is defined as an operational time interval within which more than one redundant

7.3 (Continued)

component may fail from the same cause before the first failure can be detected and corrected.

8.3 The checklist of potential common cause factors for functionally redundant equipment should be developed including consideration of (but not limited to) the following:

a) CRBRP design basis events
b) Reactor operational experiences
c) Interdependencies
d) Interfacing system initiators
e) Common locations/environments
f) Common supporting systems
g) Common procedures (human errors)
h) Single failure points effecting redundant trains (hardware and human actions)

9.3 Factors that tend to reduce susceptibility to common cause failures and should be considered during the CCF analyses are:

a) Design margins beyond design operating limits
b) Distance or event propagation barriers between redundant trains
c) Design, operation, servicing, auxiliary support, and maintenance diversity of functionally redundant equipment
d) Lack of credible (out of design limits) events that could effect the equipment.

10.3 Assessing the level of concern to associate with identified CCFs is judgemental. This judgement could be based on an estimate of the

10.3 (Continued)

probability of occurrence of events that could cross barriers between redundant trains and subject all functionally redundant components to loads in excess of their design basis. A comparison of the credibility of these common events with random failure rate data may provide a meaningful basis for judging the importance of CCFs.

11.5 The following CCF definition was proposed and agreed to by the reliability program participants at a reliability meeting at AI on February 6, 1980. CCF was defined as: "Inability of multiple first-in-line items to perform as required in a defined critical time period due to a single underlying defect or physical phenomena such that the end effect is judged to be a loss of one or more systems". (extracted from the paper by Smith/Watson, "Common Cause Failures - A Dilemma in Perspective", 1980 ARAMS).

12.6 The above definition describes the universal set of CCFs; however, it is not productive to analyse for the universal set. CCFA should only analyse for the loss of functional redundancy sub-set of the universal set of CCFs for the following reasons.

If a system function has only one functional path, it is not important whether failure in that path was multiple (from a common cause) or single (from a random cause). The system effect is the same, loss of the single functional path. These failures including dependencies should be adequately treated by FMEA.

When redundancy of functional paths exists, randomness versus commonality of cause (including common dependency) is important because the system effect is different. A failure from a random cause is highly unlikely to occur in more than one redundant path simultaneously which would leave the system functional. Failures from common causes may occur in more than one redundant path rendering the system inoperable; therefore, the project resources allocated to CCFA should be concentrated on analyzing and assessing system functional redundancies for failure susceptibility to common cause factors. This approach would significantly reduce the scope and complexity of CCF analyses and permit a more disciplined systematic and tractable approach. System schematics of redundant functions should be used to scope and guide the analyst and the reviewer from a beginning to an end of the analyses. The graphics used should be included in the assessment document for convenience of the reviewer. The graphics should provide visibility of the scope and discipline of the analysis from function, to equipment, to failure cause/mode identification, and the rationale for closeout or identification of the additional actions required.

RELIABILITY DESIGN SUPPORT DOCUMENT (RDSD)

Requirements

1.1 Reliability analyses, evaluations and dispositions for all RRCL items (including information from vendor fabrication and test activities) shall be documented in a Reliability Design Support Document (RDSD). One RDSD may document the analyses for one or several RRCL items based on logical groupings.

2.3 RDSDs shall be identified as being preliminary or final.

3.3 The equipment analyzed shall be clearly and concisely defined and described by proper nomenclature, and applicable drawing numbers including specific revisions.

4.3 The grouping of equipment for RDSD documentation shall be compatible with the RRCL.

5.3 Failure mode information resulting from qualitative reliability assessments that could alter failure rate data or the equipment operating assumptions being used in a system model shall be highlighted in the appropriate RDSD and by separate letter for consideration by the modeling organization.

6.3 The interface requirements specified for RRCL systems shall be reviewed and the failure effects on the systems function of loss of each interface shall be assessed and documented in the system RDSD.

7.3 A summary of the more significant CCFs identified shall be included in the RDSD. The basis for dispositioning each significant CCF or the planned actions should be identified in the summary.

8.4 A listing of interface inputs critical to system operation shall be provided in the system assessments.

9.4 The final resolution of concerns and uncertainties shall be provided in the RDSD for the document to be considered final.

10.4 An explicit statement shall be provided in all future RDSDs that the conclusions provided do or do not significantly alter the RSS or SHRS numerical assessments provided in WARD-D-0118 or GEFR 0007. This assessment shall be provided to the respective modeling RM.

11.5 Component failures due to an interface shall be brought forward from component analyses for assessment and documentation of their effects at the system level in RDSDs.

Guidelines

1.1 In order to achieve a degree of uniformity between RM RDSDs, the general format of attachment (2) should be utilized.

2.3 Summary results should appear near the beginning of the RDSD and should be positively written because of the positive findings or the positive actions planned and scheduled to resolve concerns or uncertainties.

3.3 Summary of system level RDSDs should include statements about the individual system reliability as well as its effect on overall SHRS or RSS systems reliability.

NUMERICAL ASSESSMENTS

Requirements

1.2 System modeling and quantification shall provide a tool to evaluate the systems' sensitivity to hardware failure rates and operating assumptions. A by-product of the modeling program shall be the overall system failure probability assessments.

2.2 RSS - A quantitative reliability assessment of the RSS failure probability has been documented in WARD-D-0118. There is currently no requirement to update this assessment. The qualitative analyses and test program shall be monitored to determine if the need for an RSS assessment update develops.

3.2 SHRS - GEFR-007 documents the current SHRS quantitative reliability assessment. The model shall be updated as results from individual RRCL item analyses become available. Sensitivity studies shall be conducted to support SHRS system functional failure probability evaluations. An interim and final update of the overall SHRS reliability assessment shall be conducted to reflect significant model revisions.

4.2 The RMs shall review the modeling failure rate assignments, operating assumptions, and failure mode coverage for their RRCL components and systems based on the qualitative insights they have gained from their RRCL item design support assessments.

5.2 Recommended model changes with supporting rationale, from the RMs model review, shall be forwarded to the modeling RM.

6.2 The SHRS modeling group shall evaluate the reliability design support qualitative assessments and make the necessary quantitative assignment changes or system assumption changes for model revisions.

REFERENCES

(1) Letter LW80462, D. K. Goeser to W. W. Dewald, et al., "CRBRP; Reliability Related Components List (RRCL) Draft Interface Control Document (ICD)", dated October 27, 1978.

(2) Letter LA80509, J. E. Nolan to R. Balent, et al., "CRBRP; Reliability Program Integration", dated December 18, 1978.

(3) Letter LG90272, D. K. Goeser to G. G. Glenn, et al., "CRBRP; Summary of Reliability Program Documentation Review", dated May 2, 1979.

(4) Letter LG90699, P. W. Dickson to G. G. Glenn, et al., "CRBRP; Summary of W-LRM Reliability Program Review with the Project Office", dated November 12, 1979.

(5) Letter LA800098, W. J. Purcell to R. Balent, et al., "CRBRP; Reliability Program Common Cause and Dependency (Interface) Analysis", letter not dated.

(6) Letter LG800269, W. J. Purcell to G. G. Glenn, et al., "CRBRP; Reliability Program, CCFA Methodology", dated May 23, 1980.

(7) Letter LW800393, W. J. Purcell to W. W. Dewald, et al., "CRBRP Reliability Program Requirement," dated October 10, 1980.

APPENDIX E

SPECIFICATION NO. FA81-002
CONSOLIDATED EDISON CO. OF NEW YORK, INC.

This Appendix represents an equipment specification which establishes general criteria for reliability requirements and guidelines for the preparation of a reliability program plan. This was prepared by Consolidated Edison as a specification for purchase of condenser circulating water pumps to improve reliability of the equipment.

E

Appendix E

SPECIFICATION NO. FA 81-002

Consolidated Edison Co. of New York, Inc.

4 Irving Place

New York, N.Y. 10003

Field and Application Engineering Department

Application Engineering Sub-Section

Description

Reliability Requirement For

Condenser Circulating Water Pumps

Location

Indian Point Station

Unit No. 2

PREPARED BY: R. A. Riscica

R. A. RISCICA - Application Engineer

APPROVED BY: E. G. Alfonsin

E. G. ALFONSIN - Chief Application Engineer

CONCURRED BY: Raymond C. Rossi

R. C. ROSSI - Reliability Engineering

DATE APPROVED: April 29, 1981

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CONSOLIDATED EDISON COMPANY OF NEW YORK

RELIABILITY REQUIREMENTS FOR

CONDENSER CIRCULATING WATER PUMPS

1.0 SCOPE

This specification establishes general criteria for reliability requirements and guidelines for the preparation and implementation of a reliability program plan for Condenser Circulating Water Pumps (hereinafter referred to as Circ Water Pumps) and their integral components purchased by Consolidated Edison Company of New York, Inc. (Con Edison).

2.0 APPLICATION

This specification, when referenced in an "Invitation to Bid," "Pump Specification," or "Contract," shall apply to Circ Water Pumps and their integral components as specified by the equipment specification.

3.0 GENERAL RELIABILITY REQUIREMENTS

General reliability requirements are set forth in the following sections.

3.1 Quantitative Reliability Requirements

The Circ Water Pumps shall comply to the values of (1) Mean Time Between Failure (MTBF) as 87,600 hours, (2) Mean Time To Repair (MTTR) as 400 hours and (3) Design Availability of 99.5% at rated design conditions of head and capacity as specified by the attached equipment specification MP 81-002. The establishment of any parameters, necessary for the determination of these reliability requirements shall be the sole responsibility of the Vendor.

3.2 Reliability Program

The Vendor shall establish and maintain an effective reliability program that is planned and developed to permit the achievement of overall program objectives, in a manner economical for Con Edison. The reliability program shall include the management and tehhnical resources, plans, procedures, schedule, and controls for the work to meet the reliability requirements. All elements of the reliability program shall be stated in the reliability program plan. Program implementation should be integrated into the design phase without impact on the schedule.

3.3 Reliability Program Plan

The Vendor shall develop and submit a reliability program plan as a separate and complete entity, separately priced, within the Vendor's proposal. The plan will describe in detail the Vendor's reliability program, which shall comply with the elements listed in Section 4.

3.4 Reliability Demonstration

Minimum acceptable equipment reliability shall be demonstrated by means of tests and/or analyses as specified in Section 4.3 of this specification.

4.0 DETAILED RELIABILITY REQUIREMENTS

The reliability requirements are covered in detail in the following subsections.

4.1 Reliability Program Plan

The Vendor's reliability program plan shall describe the methods of conducting the reliability program in order to meet the requirements of this specification. The reliability program plan shall, as a minimum, contain those elements specified in Sections 4.1.1 to 4.1.4.

4.1.1 Reliability Organization

The size, technical capabilities, organizational structure, responsibilities, and placement within the Vendor's overall company organization of the group responsible for fulfilling the reliability requirements, set forth in this document, shall be described by the Vendor.

4.1.2 Reliability Analysis

Reliability analyses of the Circ Water Pumps and their integral components shall be initiated upon receipt of a purchase order. The purpose of these analyses is to provide the basis of a reliability prediction to demonstrate compliance to the quantitative reliability requirements and to identify items critical to the safe performance of the pumps within the bounds of the design criteria specified in the equipment specification. "Critical items" will include the integral components of the pumps and maintenance/operating criteria having a significant impact upon the pumps' reliable operation. The Vendor shall establish a program for identification, control, and special handling of critical integral pump components from the design through the acceptance stage.

4.1.2 (Cont'd)

Types of reliability analyses typically performed include, but need not be limited to, any combination of the following:

- Failure Mode and Effect Analyses (FMDA)
- Fault Tree Analysis (FTA)
- Reliability modeling
- Statistical analyses for integral component failure rate determination
- Testing of circulating water pumps and/or integral components.

The Vendor, in the reliability program plan, shall indicate the type(s) of reliability analyses it intends to perform, and submit the appropriate procedures for performance of the analyses.

4.1.3 Design Reviews

Reviews shall be made at appropriate times prior to production to evaluate whether sufficient reliability requirements are included in the final design of the equipment. The planned review shall include, to the extent applicable (but shall not necessarily be limited to), (1) current reliability estimates and achievements derived from reliability analysis and/or test(s), (2) potential problem areas in design or production (derived from reliability analyses) and control measures necessary to preserve the inherent reliability, (3) corrective action on reliability critical items, and (4) effects of engineering decisions, changes, and tradeoffs upon reliability achievements, potential, and growth. The results of reliability reviews shall be documented and the Vendor shall notify Con Edison 7 days prior to each design review activity so that Con Edison personnel may attend. There shall be at least 3 design review meetings prior to production of the Circ Water Pumps. No final production pumps shall be manufactured for Con Edison usage prior to final or critical design review. In the proposal stage the milestone chart should be presented in such a manner to show sequence and time duration between each activity. Within thirty (30) days of receipt of purchase order, the chosen vendor shall submit its detailed program milestone chart with dates.

4.1.4 Program Milestones

The reliability program plan shall include a milestone chart of all the reliability activities specified in the plan. This includes schedules for testing, design reviews, reliability analysis, etc. The milestone chart shall be updated as needed to reflect the current schedule.

4.2 DOCUMENTATION

A report shall be provided to Con Edison whthin two (2) weeks after each design review activity. This report will contain the following:

1. Design review minutes

2. Reliability analysis to date

3. Tradeoffs necessary to meet the reliability requirements

4. Results of all demonstration tests

Additionally, seven (7) working days prior to each design review meeting the Vendor shall identify and if necessary supply the latest revisions of drawings and parts lists.

Monthly progress reports shall also be submitted updating the results of the reliability effort.

A final reliability report shall be provided to Con Edison within two (2) weeks after delivery of the Circ Water Pumps. This report will summarize the results of the reliability effort to date, show all demonstration test results, and include the final reliability prediction for the Circ Water Pumps.

4.3.0 Reliability Demonstration Requirements

Reliability demonstration requirements by the Vendor of the Circ Water Pumps are described in the folloiing subsection.

4.3.1 Reliability Prediction

A reliability prediction, based on use of known sources of generic and/or experience failure data for elements of the pump(s) shall be prepared. The availability of the pump(s) resulting from this prediction shall be equal to or greater than specified in the technical specifications. (Note - the MTBF and MTTR may vary from the values specified to provide design tradeoffs so long as the specified availability is maintained.

4.3.2 Shop Drawings

Within two weeks of delivery, the vendor shall supply detailed shop drawings to Con Edison. This is to provide assurance for continued availability and timely delivery of spare parts throughout the expected equipment life.

4.4.0 Warranty

The warranties set forth in Con Edison's Standard Terms and Conditions for Purchase of Major Equipment shall be applicable to condenser circulating water pumps and its components for a period of five (5) years from the date of the initial operation of the pump(s) in normal service. If at any time the circulating water pump(s) cannot provide the required capacities and head, as a system, a determination will be made by Con Edison as to which pump(s) has degraded to a point in which a plant derating is feasible and will invoke the warrantee for those pumps.

5.0 APPENDIX

Appendix A has definitions useful to the Vendor.

APPENDIX A

DEFINITIONS

MEAN-TIME-BETWEEN-FAILURES-(MTBF) — For a particular interval, the total functioning life of a population of an item divided by the total number of failures within the population during the measurement interval. The definition holds for time, cycles, events, or other measure of life units.

MEAN-TIME-TO-REPAIR (MTTR) — The total corrective maintenance time divided by the total number of corrective maintenance actions during a given period of time. (Given that qualified personnel, and necessary spare parts and repair equipment are available.)

DESIGN AVAILABILITY (A) — A measure of the degree to which an item is in the operable and committable state when called upon for service at an unknown or random point in time.

The relationship to be used to express availability involves the Mean-Time-Between-Failures (MTBF) and the Mean-Time-To-Repair (MTTR), as shown below.

$$A = \frac{MTBF}{MTBF + MTTR}$$

APPENDIX F

OAK RIDGE NATIONAL LABORATORY (ORNL)
NATIONAL SAFETY INFORMATION CENTER

PRA/RELIABILITY BIBLIOGRAPHY

This appendix presents the results of a bibliographic search of the files of the National Safety Information Center of ORNL. The search incompassed those technical papers, reports, etc., concerned with the areas of risk, reliability and probabilistic risk assessment (PRA).

F

0/C000001-0000077//   1

SSION NO.   00C0169704
LE   ANALYSIS OF FILTERED VENTING AS A MITIGATION SAFETY FEATURE IN PWRS WITH LARGE, DRY CONTAINMENTS
HOR(S)   SCHECHTMAN R;HEISING C
PAUTH   MASS. INST. OF TECHNOLOGY, CAMBRIDGE
E   1981
PE   L
D   9 PPS. FROM ANS/ENS TOPICAL MEETING ON PROBABILISTIC RISK ASSESSMENT; PORT CHESTER, NY, SEPT. 1981
IL   AVAILABILITY - CAROLYN HEISING, ASSISTANT PROFESSOR, NUCLEAR ENGINEERING DEPT., MASS. INST. OF TECHNOLOGY, CAMBRIDGE, MA 02139
EGORY   120000;110000;230000
TION   0135
P CODE   MDM
NTRY   A
TRACT   THIS PAPER DELINEATES A METHOD FOR STRUCTURING THE ANALYSIS OF THE RISK REDUCTION POTENTIAL OF A FILTERED VENTING CONTAINMENT SYSTEM (FVCS). EVENT TREES ARE DEVELOPED INDICATING THE IMPACT ON ACCIDENT SEQUENCE DEFINITION OF THE INTRODUCTION OF A FVCS, PARTICULARLY FOR THE CASE OF A LARGE, DRY, PWR CONTAINMENT. WORK HAS BEEN DONE TO QUALITATIVELY COMPARE THE RELATIVE RISKS OF THESE NEWLY INTRODUCED ACCIDENT SEQUENCES WITH THE POTENTIAL FOR RISK REDUCTION OF THE STANDARD PWR ACCIDENT SEQUENCES. (EWH)
WORDS   REACTOR, PWR;CONTAINMENT, PRESSURE VENTING;PROBABILITY;RISK; SAFETY ANALYSIS;SAFETY EVALUATION

/0/0000001-0000077//   2
SSION NO.   00X0169665
LE   PROGRESS OF NUCLEAR SAFETY RESEARCH PART I (REACTOR SAFETY RESEARCH)
PAUTH   JAPAN ATOMIC ENERGY RESEARCH INST., TOKAI
    1980
    N
D   JPNRSR-270 +. 120 PPS. TABS, FIGS, 1980
L   AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
GORY   110000;130000;170000
ION   0135
CODE   JAE
TRY   J
RACT   THIS PAMPHLET PRESENTS THE ACTIVITIES ASSOCIATED WITH THE REACTOR SAFETY RESEARCH WHICH WAS CONDUCTED BY JAERI, ESPECIALLY HIGHLIGHTING THE ACTIVITIES CARRIED OUT DURING THE PAST TWO YEARS. THIS PRESENTATION PUTS SPECIAL EMPHASIS ON THE RESEARCH ACTIVITIES CARRIED OUT AT THE REACTOR SAFETY RESEARCH CENTER AND ALSO INCLUDES THE INTERNATIONAL COOPERATIVE RESEARCH PROGRAMS IN WHICH JAERI IS A PARTICIPANT. FOR EDITORIAL CONVENIENCE, THIS PAMPHLET CONSISTS OF TWO PARTS: PART I DESCRIBES THE REACTOR SAFETY RESEARCH AND PART II THE ENVIRONMENTAL SAFETY RESEARCH. (FAH)
ORDS   JAPAN;FUEL CYCLE;EXAMINATION;FRACTURE TOUGHNESS;PIPE WHIP; CORROSION;CRACK;PROBABILITY;RISK;FOREIGN EXCHANGE;ENGINEERED SAFETY FEATURE;R AND D PROGRAM

0/0000001-0000077//   3
SSION NO.   0080169678
E   THE APPLICATION OF PROBABILISTIC RISK ASSESSMENT TECHNIQUES TO ENERGY TECHNOLOGIES
R(S)   RASMUSSEN NC
UTH   MASS. INST. OF TECHNOLOGY, CAMBRIDGE
    1981
    J
    16 PPS, 4 FIGS, 12 REFS. ANNUAL REVIEW OF ENERGY, VOL. 6, PP. 123-38 (1981)
ORY   230000
ON   0135
CODE   MBM
RY   A
ACT   DURING THE LAST DECADE THERE HAS BEEN A RAPID EXPANSION IN THE DEVELOPMENT OF TECHNIQUES FOR CARRYING OUT PRA'S ON A WIDE VARIETY OF ACTIVITIES, AND ALSO IN THE NUMBER OF PEOPLE TRAINED

IN THE USE OF THESE TECHNIQUES. THE METHODS FOR ESTIMATING EQUIPMENT FAILURE PROBABILITY ARE QUITE WELL DEVELOPED. THE DOMINANT CONTRIBUTOR TO THE UNCERTAINTY IN THE RESULTS IS THE UNCERTAINTY IN FAILURE RATES OF COMPONENTS AND IN THE HUMAN ERROR RATES OF OPERATORS AND MAINTENANCE PERSONNEL. THE TECHNIQUES FOR COMPARING RISKS OF DIFFERENT TYPES NEED TO BE MORE FULLY DEVELOPED IF WIDELY ACCEPTED WAYS OF HANDLING THIS DIFFICULT PROBLEM ARE TO BE ACHIEVED.

WORDS PROBABILITY;RISK;ACCIDENT, PROBABILITY OF;FAILURE, EQUIPMENT; FAILURE, COMPONENT;HUMAN FACTORS;BENEFIT VS RISK;PROBABILISTIC RISK ASSESSMENT

/0/0000001-00000777/ 4

CESSION NO. 00C0169529

LE GENERIC NUCLEAR SAFETY ISSUES: METHODS AND EXAMPLES

HOR(S) HEISING CD

PAUTH MASS. INST. OF TECHNOLOGY, CAMBRIDGE

E 1981

E L

U 10 PPS, FROM ANS/ENS TOPICAL MEETING ON PROBABILISTIC RISK ASSESSMENT; PORT CHESTER, NY, SEPT. 1981

IL AVAILABILITY - CAROLYN D. HEISING, ASSISTANT PROFESSOR, NUCLEAR ENGINEERING DEPT., MASS. INST. OF TECHNOLOGY, CAMBRIDGE, MA 02139

EGORY 010000;230000;180000

TION 0135

P CODE MGM

NTRY A

TRACT METHODS FOR ANALYZING GENERIC NUCLEAR SAFETY ISSUES RELATED TO NUCLEAR POWER PLANT LICENSING ARE DESCRIBED. AN EXAMPLE IS GIVEN OF A PARTICULAR APPLICATION TO ESTIMATING THE REACTOR CORE MELT FREQUENCY INCLUDING THE EXPERIENCE GAINED AT THREE MILE ISLAND. USING A BAYESIAN APPROACH, THE RESULTS INDICATE THAT COMMERCIAL REACTOR OPERATING EXPERIENCE TO DATE WHICH ACCOUNTS FOR THE POSSIBILITY OF CORE MELTDOWNS AT BROWNS FERRY AND THREE MILE ISLAND DOES NOT SHIFT THE ESTIMATE OF WASH-1400 SIGNIFICANTLY. (EWH)

WORDS PROBABILITY;RISK;SAFETY ANALYSIS;SAFETY EVALUATION;GENERIC; ANALYTICAL TECHNIQUE

/0/0000001-00000777/ 5

SSION NO. 00C0169176

E OCONEE PROBABILISTIC RISK ASSESSMENT: METHODOLOGY, APPLICATIONS, AND EXPERIENCE

HOR(S) LEWIS SR;SUGNET WR

PAUTH DUKE POWER CO., CHARLOTTE, NC - EPRI NUCLEAR SAFETY ANALYSIS CENTER, PALO ALTO, CA

1981

L

11 PPS, FROM ANS/ENS TOPICAL MEETING ON PROBABILISTIC RISK ASSESSMENT; PORT CHESTER, NY, SEPT. 1981

L AVAILABILITY - WILLIAM R. SUGNET, EPRI NUCLEAR SAFETY ANALYSIS CENTER, PALO ALTO, CA 94303

GORY 230000;090000;170000

ION 0134

CODE APY;EPR

TRY A

RACT A JOINT EFFORT BY THE NUCLEAR SAFETY ANALYSIS CENTER (NSAC), DUKE POWER COMPANY, A NUMBER OF OTHER UTILITY COMPANIES, AND THE INSTITUTE OF NUCLEAR POWER OPERATIONS (INPO) HAS BEEN UNDERTAKEN TO PERFORM AN INTEGRATED PROBABILISTIC RISK ASSESSMENT OF DUKE'S OCONEE NUCLEAR STATION UNIT 3. STATE-OF-THE-ART METHODS WERE REVIEWED AND A METHOD SELECTED FOR EACH OF THE MAJOR TASK AREAS OF THE ANALYSIS. THE METHODS CHOSEN AND THEIR APPLICATION ARE DISCUSSED, WITH EMPHASIS ON AVAILABLE RESULTS TO DATE AND EXPERIENCE THAT CAN BE APPLIED TO FUTURE UTILITY AND INDUSTRY STUDIES OF THIS TYPE. (EWH)

ORDS OCONEE 3 (PWR);REACTOR, PWR;PROBABILITY;RISK;ANALYTICAL TECHNIQUE;RELIABILITY, SYSTEM;FAULT TREE ANALYSIS;PROBABILISTIC RISK ASSESSMENT

/0000001-00000777/ 6

SSION NO. 00C0168977

E UNCERTAINTITY IN PROBABILISTIC RISK ANALYSIS

HOR(S) PARRY GW;TEAGUE HJ;WINTER PW
PAUTH UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, U.K.
E 1981
E L
O STI/PUB/566 (VOL. II) +. 17 PPS. PP. 487-503 OF PROCEEDINGS OF AN INTERNATIONAL CONFERENCE ON CURRENT NUCLEAR POWER PLANT SAFETY ISSUES; STOCKHOLM, SWEDEN, OCT. 20-24, 1980
AIL AVAILABILITY - UNIPUB, INC., P.O. BOX 433, NEW YORK, N.Y. 10016
EGORY 230000
TION 0134
P CODE UKA
NTRY W
TRACT THE SOURCES OF UNCERTAINTY IN PROBABILISTIC RISK ANALYSIS ARE DISCUSSED USING THE EVENT/FAULT TREE METHODOLOGY AS AN EXAMPLE. THE ROLE OF STATISTICS IN QUANTIFYING THESE UNCERTAINTIES IS INVESTIGATED. IT IS ARGUED THAT BAYESIAN STATISTICS IS A MORE APPROPRIATE VEHICLE FOR THE PROBABILISTIC ANALYSIS OF RARE EVENTS THAN CLASSICAL STATISTICS, AND A SHORT REVIEW IS GIVEN.
WORDS UNITED KINGDOM;PROBABILITY;RISK;STATISTICAL ANALYSIS;POWER PLANT, NUCLEAR;FAULT TREE ANALYSIS

0/0000001-0000077// 7
ESSION NO. 0000168976
LE DETERMINATION OF QC REQUIREMENTS FOR PROBABILISTIC RISK ASSESSMENT
HOR(S) DECKERS J
PAUTH TECHNISCHER UBERWACHUNGS-VEREIN RHEINLAND, F.R. GERMANY
E 1981
E L
STI/PUB/566 (VOL. II) +. 12 PPS. PP. 383-94 OF PROCEEDINGS OF AN INTERNATIONAL CONFERENCE ON CURRENT NUCLEAR POWER PLANT SAFETY ISSUES; STOCKHOLM, SWEDEN, OCT. 20-24, 1980
L AVAILABILITY - UNIPUB, INC., P.O. BOX 433, NEW YORK, N.Y. 10016
EGORY 230000;010000
TION 0134
NTRY W
TRACT A METHODOLOGY IS SHOWN FOR THE PROBABILISTIC FAILURE ANALYSIS OF PRESSURE RETAINING COMPONENTS. THE PURPOSE IS TO FIND, TO APPRAISE, AND TO DEFINE BOUNDARY CONDITIONS FOR THE QUALITY REQUIREMENTS, MEASURES OF QUALITY CONTROL, AND CRITERIA FOR INSPECTION AND TESTING. THE FAILURE PROBABILITY OF T E REACTOR PRESSURE VESSEL WAS DETERMINED USING A MONTE CARLO CODE. THE RESULTS SHOW THE MAGNITUDE OF THE INFLUENCE OF THE DIFFERENT PARAMETERS SUCH AS TOUGHNESS, NEUTRON FLUX, DEFECT SIZE, AND DISTRIBUTION, LENGTH-TO-DEPTH RATIO.
WORDS GERMANY;QUALITY ASSURANCE;PROBABILITY;RISK;MONTE CARLO;CRACK; PRESSURE VESSELS;PIPES AND PIPE FITTINGS;REACTOR, PWR

0/0000001-0000077// 8
ESSION NO. 0000168975
LE IDENTIFICATION AND EVALUATION OF ACCIDENT SEQUENCES IN NUCLEAR POWER REACTORS
HOR(S) AMENDOLA A;OLIVI L;REINA G
PAUTH JOINT RESEARCH CENTRE, ISPRA ESTABLISHMENT, ISPRA, ITALY ; MESA, MILAN, ITALY
1981
L
STI/PUB/566 (VOL. II) +. 21 PPS. PP. 343-63 OF PROCEEDINGS OF AN INTERNATIONAL CONFERENCE ON CURRENT NUCLEAR POWER PLANT SAFETY ISSUES; STOCKHOLM, SWEDEN, OCT. 20-24, 1980
L AVAILABILITY - UNIPUB, INC., P.O. BOX 433, NEW YORK, N.Y. 10016
EGORY 230000
TION 0134
NTRY W
TRACT PROBABILISTIC ANALYSIS TECHNIQUES ARE BEING USED MORE AND MORE USED FOR THE EVALUATION OF ACCIDENT PROGRESSION IN NUCLEAR POWER PLANTS. THE PAPER REVIEWS THE CONTRIBUTIONS IN PROGRESS AT JRC-ISPRA, SPECIFICALLY REPORTS ON THE FOLLOWING: 1) THE SET-UP OF A EUROPEAN RELIABILITY DATA SYSTEM FOR THE ACQUISITION AND ORGANIZATION OF OPERATIONAL DATA OF LWRS IN THE EUROPEAN COMMUNITY, 2) THE DEVELOPMENT OF MORE COMPLETE AND REALISTICS MODELS OF SYSTEMS, AND 3) THE DEVELOPMENT OF RESPONSE SURFACE METHODOLOGY FOR ANALYSIS OF UNCERTAINTY PROPAGATIONS IN CONSEQUENCE AND IN PROBABILITY OF ACCIDENT

CHAINS.
WORDS   ITALY;PROBABILITY;RISK;FAILURE, SEQUENTIAL;EUROPE;DATA COLLECTION;MODEL

/0/0000001-0000077//   9
ESSION NO.   00C0168971
LE   RISK ASSESSMENT
HOR(S)   ECKERED T;OKRENT D
E   1981
E   L
D   STI/PUB/566 (VOL. I) +. 2 PPS, PP. 498, 499 OF PROCEEDINGS OF AN INTERNATIONAL CONFERENCE ON CURRENT NUCLEAR POWER PLANT SAFETY ISSUES; STOCKHOLM, SWEDEN. OCT. 20-24, 1980
IL   AVAILABILITY - UNIPUB, INC., P.O. BOX 433, NEW YORK, N.Y. 10016
EGORY   230000
TION   0134
NTRY   A
TRACT   THE CHAIRMAN SUMMARIZED THE PAPERS ON RISK ASSESSMENT. THE EIGHT PAPERS PRESENTED IN THE FIRST SESSION GAVE A PICTURE OF THE PRESENT STATUS OF RISK ASSESSMENT TECHNIQUES AND USES. THE SECOND SESSION CONTAINED A TOTAL OF NINE PAPERS. ONE PAPER FROM THE USSR DISCUSSED RELIABILITY EXPERIENCE WITH THE VVER-440 REACTORS. ONE PAPER FROM THE UK DISCUSSED SOME OF THE UNCERTAINTIES WHICH ENTER INTO PROBABILISTIC RISK ANALYSIS. OTHER PAPERS ILLUSTRATED HOW PROBABILISTIC METHODOLOGY IS CURRENTLY BEING USED IN DESIGN AND HOW IT IS GRADUALLY ENTERING INTO THE LICENSING PROCESS.
WORDS   RISK;PROBABILITY;LICENSING PROCESS;DESIGN;RELIABILITY ANALYSIS

/0/0000001-0000077//   10
ESSION NO.   00E0168230
E   PROBABILISTIC MODELS FOR THE BEHAVIOR OF COMPARTMENT FIRES
HOR(S)   SIN NG
AUTH   UNIV. OF CALIF., LOS ANGELES
             1981
             H
             NUREG/CR-2269 +. 160 PPS, FIGS, REFS, AUG. 1981
L   AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GORY   230000;100000;090000
ION   0132
 CODE   UAV
TRY   A
RACT   PHYSICAL MODELS WHICH PREDICT THE THERMAL HAZARDS (INCLUDING TEMPERATURES AND HEAT FLUXES) DURING A COMPARTMENT FIRE AS FUNCTIONS OF SPACE AND TIME ARE DEVELOPED. SINCE LARGE UNCERTAINTIES ARE INHERENT TO THE ANALYSIS, THE MODELS ARE PROBABILISTIC. GENERAL MODELS ARE CONSTRUCTED FOR THE PERIODS OF FIRE GROWTH AND FULLY-DEVELOPED BURNING. THESE MODELS ARE USED IN SAMPLE ANALYSES TO ESTIMATE THE FIRE HAZARD IN PARTICULAR COMPARTMENTS. THE OVERALL METHODOLOGY REQUIRES THE SYNTHESIS OF A DETERMINISTIC PHYSICAL MODEL FROM INFORMATION AVAILABLE IN THE LITERATURE.
ORDS   FIRE;MODEL, PHYSICAL;PROBABILITY;COMPARTMENT;BEHAVIOR;RISK;HEAT TRANSFER, CONVECTION;IGNITION TEMPERATURE;CABLES AND CONNECTORS; HEAT TRANSFER, RADIANT;HEAT TRANSFER, RADIANT

0/0000001-0000077//   11
SSION NO.   00X0167677
E   ISSUES AND PROBLEMS IN INFERRING A LEVEL OF ACCEPTABLE RISK
OR(S)   SALEM SL;SOLOMON KA;YESLEY MS
AUTH   THE RAND CORP., SANTA MONICA, CA
             1980
             N
             R-2561-DOE +. 110 PPS, 13 TABS, 5 FIGS, REFS, AUG. 1980
             AVAILABILITY - PUBLICATIONS DEPT., THE RAND CORP., 1700 MAIN ST., SANTA MONICA, CALIF. 90406
GORY   230000
ION   0132
CODE   AAN
RY   A
ACT   THIS REPORT PRESENTS THE FINDINGS FROM A REVIEW, OF HOW TECHNICAL AND NONTECHNICAL FACTORS INFLUENCE THE PERCEPTION AND REGULATION OF RISKS FROM TECHNOLOGICAL ACTIVITIES. THE STUDY

CONSIDERS: HOW EXISTING LEVELS OF RISK VARY ACROSS DIFFERENT INDUSTRIES AND HAZARDS. HOW VIEWPOINTS AND DEFINITIONS OF ACCEPTABLE RISK DIFFER. WHAT ONE CAN INFER ABOUT LEVELS OF ACCEPTABLE RISK FOR CURRENT AND EMERGING ENERGY TECHNOLOGIES. WHAT THE IMPLICATIONS ARE FOR THE REGULATION OF CURRENT AND FUTURE RISKS TO SOCIETY. THE REPORT IS ADDRESSED TO POLICY ANALYSTS, REGULATORS, LEGISLATORS, AND OTHERS CONCERNED WITH SAFETY AND RISK-REDUCTION PROGRAMS FOR EXISTING AND EMERGING TECHNOLOGIES.

WORDS RISK;BENEFIT VS RISK;HAZARDS ANALYSIS;PROBABILITY; SOCIO/PHILOSOPHICAL CONSIDERATION;HAZARD, RELATIVE;ACCIDENT, PROBABILITY OF

/0/0000001-00000777/ 12

ESSION NO. 00E0167675

LE RACAP-1: REACTOR ACCIDENT CONSEQUENCE ANALYSIS PROGRAM - FIRST VERSION. VOLUME 1: THEORY AND METHODS

HOR(S) BAILEY PG;DENNY VE;STEVENSON DE

E 1981

E R

) EPRI-NP-1871 (VOL. 1) +. 225 PPS, MAY 1981

IL AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10090, PALO ALTO, CALIF. 94303

EGORY 050000;230000

TION 0132

CODE SAI

NTRY A

RACT THIS VOLUME DOCUMENTS THE GENERAL SCOPE AND HISTORICAL DEVELOPMENT OF CONSEQUENCES ANALYSIS FOR RISK ASSESSMENT PURPOSES. IT OUTLINES THE REACTOR ACCIDENT CONSEQUENCE ANALYSIS PROGRAM (RACAP) NETWORK OF SIX COMPUTER CODES THAT CAN BE USED TO PERFORM INTEGRATED ANALYSES OF PLANT PHYSICAL RESPONSE, RADIOACTIVITY BEHAVIOR, AND EXPLANT CONSEQUENCES FOR POSTULATED ACCIDENT SEQUENCES. THE MODELING DETAILS AND LOGIC STRUCTURE OF EACH CODE IN THE RACAP NETWORK ARE DESCRIBED IN SEPARATE SECTIONS OF THIS VOLUME. A SECTION OUTLINING POTENTIALFUTURE ADDITIONS AND IMPROVEMENTS IN THE METHODOLOGY IS ALSO INCLUDED.

ORDS COMPUTER PROGRAM;RISK;ACCIDENT, CONSEQUENCES;REACTOR PHYSICS; PROBABILITY;ACCIDENT, LOSS OF COOLANT;ACCIDENT, CORE DISRUPTIVE; CONTAINMENT;EPRI;CORE MELTDOWN

0/0000001-00000777/ 13

SSION NO. 0020167383

E CPC PROBABILISTIC RISK ASSESSMENT FOR BIG ROCK POINT

AUTH CONSUMERS POWER CO., JACKSON, MI

1981

W

APPROX. 650 PGS, LTR W/ATTACH. TO NRC DIRECTOR OF NUCLEAR REACTOR REGULATION, APR 6, 1981, DOCKET 50-155, TYPE--BWR, MFG--GE, AE--BECH. DCS NO.--8104160590

L AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)

GORY 170000;230000

ION 0131

CODE CPC

TRY A

RACT PROVIDES THE COMPLETED BIG ROCK POINT PROBABILISTIC RISK ASSESSMENT (PRA). SECTION 1, EXECUTIVE SUMMARY PROVIDES ADDITIONAL INFORMATION ON WHY THIS WORK WAS PERFORMED, HOW IT WAS PERFORMED AND WHAT RESULTS WERE OBTAINED. THE PRA INDICATES CERTAIN CONTRIBUTIONS TO RISK FOR WHICH MODIFICATIONS TO THE CURRENT PLANT DESIGN AND OPERATION WHICH WILL SUBSTANTIALLY REDUCE RISK TO THE PUBLIC. AN EVALUATION OF THESE PLANT MODIFICATIONS HAS BEEN PERFORMED BY CONSUMERS POWER COMPANY AND CONTRACTORS AS SECTION 7, EVALUATION OF PLANT MODIFICATIONS.

RDS BIG ROCK POINT (BWR);REACTOR, BWR;PROBABILITY;SAFETY ANALYSIS; HAZARDS ANALYSIS;SYSTEM ANALYSIS;RISK

/0000001-00000777/ 14

SION NO. 0030166770

FIDUCIAL ESTIMATION OF PROBABILITIES FOR RELIABILITY AND RISK

ASSESSMENT
E 1980
E U
U 2 PPS, 1 TAB, ANNALS OF NUCLEAR ENERGY, VOL. 8, PP. 147-8 (1980)
EGORY 230000
TION 0130
NTRY U
DL ANSE
TRACT THIS IS A LETTER TO THE EDITOR DETAILING ONE METHOD OF CALCULATION OF PROBABILITY OF RISK.
WORDS RISK;PROBABILITY;RELIABILITY ANALYSIS;POWER PLANT, NUCLEAR; MATHEMATICAL TREATMENT

/0/0000001-00000777/ 15
ESSION NO. 00J0166763
LE CHARACTERIZATION AND EVALUATION OF UNCERTAINTY IN PROBABILISTIC RISK ANALYSIS
HOR(S) WINTER PW;PARRY GW
PAUTH UKAEA SAFETY & RELIABILITY DIRECTORATE, U.K.
E 1981
E U
U 15 PPS, 1 FIG, 99 REFS, NUCLEAR SAFETY, 22(1), PP. 26-42 (JAN.-FEB. 1981)
EGORY 230000
TION 0130
P CODE UKA
NTRY A
DB NUSA
TRACT THE SOURCES OF UNCERTAINTY IN PROBABILISTIC RISK ANALYSIS ARE DISCUSSED, USING THE EVENT AND FAULT-TREE METHODOLOGY AS AN EXAMPLE. THE ROLE OF STATISTICS IN QUANTIFYING THESE UNCERTAINTIES IS INVESTIGATED. A CLASS OF UNCERTAINTIES IS IDENTIFIED WHICH ARE, AT PRESENT, UNQUANTIFIABLE USING EITHER CLASSICAL OR BAYESIAN STATISTICS. IT IS ARGUED THAT BAYESIAN STATISTICS IS THE MORE APPROPRIATE VEHICLE FOR THE PROBABILISTIC ANALYSIS OF RARE EVENTS, AND A SHORT REVIEW IS GIVEN WITH SOME DISCUSSION ON THE REPRESENTATION OF IGNORANCE.
WORDS RISK;PROBABILITY;FAULT TREE ANALYSIS;HAZARDS ANALYSIS; STATISTICAL ANALYSIS

/0/0000001-00000777/ 16
SSION NO. 00C0168003
LE A DESIGN TO ENHANCE DETECTION OF PRESELECTED EVENTS FOR RISK REDUCTION
HOR(S) SPURGIN AJ;TEDT FW
PAUTH GENERAL ATOMIC CO., SAN DIEGO, CA
1981
L
3 PPS, IEEE TRANS. NUCL. SCI., 28(1), PP. 902-4 (FEB. 1981)
GORY 090000;170000;230000
ION 0126
CODE LAW
TRY A
RACT THE POINT IS MADE THAT THE MENTAL FLEXIBILITY OF THE OPERATOR, WHICH IS ESSENTIAL DURING AN UNANTICIPATED ACCIDENT SEQUENCE, MUST BE IN TOP CONDITION DURING A HIGH STRESS INCIDENT - EXACTLY THE TYPE OF INCIDENT WHICH WILL PROMOTE THE MENTAL RIGIDITY. INCREASED OPERATOR TRAINING ALONE IS JUDGED TO BE INSUFFICIENT UNLESS SUPPLEMENTED BY OTHER APPROACHES TO THE PROBLEM. A SYSTEM PROPOSED BY GENERAL ATOMIC TO IMPROVE THE SITUATION IS DISCUSSED - ITS PRINCIPLES ARE GIVEN AND A TEST OF THE SYSTEM, BASED UPON THE THREE MILE ISLAND ACCIDENT, IS DESCRIBED.
ORDS FAILURE, OPERATOR ERROR;HUMAN FACTORS;TRAINING;PROBABILITY;RISK; OPERATOR ACTION

0/0000001-00000777/ 17
SSION NO. 00E0165659
E DIFFRACTION OF SHOCK WAVES ANALYSIS OF EXPERIMENTALLY ASCERTAINED SHOCK WAVES ON REACTOR BUILDINGS-FINAL REPORT (IN GERMAN)
R(S) HOFMANN H;HUBER A;FAASS E
AUTH STATIK, DYNAMIK UND KONSTRUKTION GMBH, F.R. GERMANY
1979

H
SDK REPORT 2/11 + RS 286 + GERRSR-564 +. 43 PPS. 15 FIGS. 9 REFS. MAY 1979
GUAGE GERMAN
IL AVAILABILITY - SUSAN DISILVESTRO, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL. U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
EGORY 020000
TION 0128
NTRY G
TRACT THE APPLICABILITY OF PRESENT ANALYTICAL METHODS TO PROBLEMS CONCERNING THE REFLECTION AND DIFFRACTION OF SHOCK WAVES IN THE VICINITY OF REACTOR BUILDINGS WAS INVESTIGATED. THE ANALYTICAL RESULTS WERE COMPARED WITH EXPERIMENTAL VALUES. A DISCRETE REPRESENTATION OF THE LINEAR WAVE EQUATION LEADS TO PROPER ANALYTICAL RESULTS. APPROXIMATE SOLUTIONS CAN BE GAINED FOR SOME NON-LINEAR PROBLEMS. WAVE PROPAGATION PHENOMENA IN CONNECTION WITH COMPLEX GEOMETRICAL CONFIGURATIONS (LOCATION OF REACTOR BUILDINGS) CAN NOW BE TREATED ANALYTICALLY. THE SHOCK SIZE LIMITS THE APPLICABILITY OF THE LINEAR WAVE EQUATION. THE THERMODYNAMIC NON-LINEARITIES AND THE FLOW PHENOMENA GAIN INFLUENCE WITH GREATER SHOCK SIZE.
WORDS SITING;EARTHQUAKE;EARTHQUAKE ENGINEERING;BUILDING;POWER PLANT, NUCLEAR;EARTHQUAKE PREDICTION;THEORETICAL INVESTIGATION;MODEL; PROBABILITY;WAVE, STRESS;SHOCK WAVE;RISK;GERMANY;FOREIGN EXCHANGE

0/0000001-00000777/ 18
SSION NO. 00E0165566
LE REACTOR SAFETY STUDY METHODOLOGY APPLICATIONS PROGRAM: SEQUOYAH #1 PWR POWER PLANT
HOR(S) CARLSON DD;HICKMAN JW;WOOTON R
PAUTH SANDIA NATIONAL LABS., ALBUQUERQUE, N.M. ; BATTELLE COLUMBUS LABS., OHIO
1981
H
NUREG/CR-1659 + SAND80-1897 +. APPROX. 500 PPS. FEB. 1981
L AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GORY 180000;170000
TION 0127
CODE AUA;CAF
TRY A
RACT THIS REPORT DESCRIBES WORK DONE ON THE REACTOR SAFETY STUDY METHODOLOGY APPLICATIONS PROGRAM. THE ACCIDENT SEQUENCES WHICH DOMINATE RISK HAVE BEEN IDENTIFIED FOR THE SEQUOYAH NO. 1 PRESSURIZED WATER REACTOR (PWR) POWER PLANT. A COMPARISON OF THE SYSTEMS AND ACCIDENT SEQUENCES IS MADE BETWEEN THE SEQUOYAH PLANT AND THE PWR PLANT USED IN THE REACTOR SAFETY STUDY.
URDS SYSTEM ANALYSIS;SEQUOYAH 1 (PWR);REACTOR, PWR;ACCIDENT, LOSS OF COOLANT;FAULT TREE ANALYSIS;TRANSIENT;CONTAINMENT, ICE CONDENSER;RISK;PROBABILITY;SAFETY PROGRAM;HJCK;NRC-AN

0/0000001-00000777/ 19
SSION NO. 00C0165568
E THE EXPANDING ROLE OF QUANTITATIVE RISK ANALYSIS IN THE FEDERAL REPUBLIC OF GERMANY
OR(S) BIRKHOFER A
AUTH TECH. UNIV. MUNCHEN, GARCHING, F.R. GERMANY
1980
L
IAEA-CN-39/6.5 +. 13 PPS, FROM IAEA INTERNATIONAL CONFERENCE ON CURRENT NUCLEAR POWER PLANT SAFETY ISSUES; STOCKHOLM, SWEDEN, OCT. 20-24, 1980
AVAILABILITY - UNIPUB, INC., P.O. BOX 433, NEW YORK, N.Y. 10016
GORY 010000;230000
ION 0127
CODE MIT
TRY W
ACT THIS PAPER DISCUSSES MAINLY THE PRESENT ROLE OF BOTH RELIABILITY AND RISK ANALYSIS IN NUCLEAR SAFETY. SIMPLIFICATIONS OF ANALYTICAL TOOLS, AND UNCERTAINTIES OF INPUT DATA RESULT IN CONSIDERABLE UNCERTAINTY MARGINS OF THE CALCULATED FIGURES. IN MANY CASES VALUABLE INSIGHT CAN BE

GAINED FROM RELATIVE EVALUATIONS. ABSOLUTE RELIABILITY FIGURES SHOULD BE USED CAUTIOUSLY. IF RELIABILITY REQUIREMENTS ARE TO BE FORMALLY FIXED, THEY SHOULD PREFERABLY BE RELATED TO POSTULATED ACCIDENT SEQUENCES RATHER THAN TO INDIVIDUAL SYSTEM FUNCTIONS. (EWH)

WORDS RISK;PROBABILITY;RELIABILITY ANALYSIS;GERMANY;LICENSING PROCESS; SAFETY EVALUATION;REVIEW

/0/0000001-00007777 20

ESSION NO. 00E0165249

LE OCONEE PROBABILISTIC RISK ASSESSMENT PROJECT PLAN

HOR(S) SUGNET WR;RUBBLE ET;CANADY KS

PAUTH EPRI NUCLEAR SAFETY ANALYSIS CENTER, PALO ALTO, CALIF.

C 1980

E R

O NSAC-7 +. 55 PPS. NOV. 1980

IL AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10090, PALO ALTO, CALIF. 94303

EGORY 010000;230000

TION 0127

P CODE EPR

NTRY A

TRACT A DETAILED PLAN HAS BEEN DEVELOPED DESCRIBING THE SCOPE, MAJOR TASKS AND LEVELS OF EFFORT REQUIRED FOR AN INDUSTRY TEAM TO CONDUCT A COMPREHENSIVE, BALANCED PROBABILISTIC RISK ASSESSMENT (PRA) OF DUKE POWER'S OCONEE UNIT 3. THIS PROJECT HAS BEEN INITIATED BY THE NUCLEAR SAFETY ANALYSIS CENTER WITH THE OBJECTIVE OF DIRECTLY INVOLVING NUCLEAR UTILITY ENGINEERS IN WORKING WITH PRA METHODS, AND PROVIDING A MODEL PLANS SPECIFIC STUDY TO GUIDE FUTURE SUCH EFFORTS BY NUCLEAR UTILITIES. THE PROJECT SPANS A CALENDAR YEAR AND CALLS FOR INVESTMENT OF ABOUT 12 NSAC/UTILITY MAN YEARS, ABOUT 5 CONTRACTOR MAN YEARS, AND ADDITIONAL SUPPORT RESOURCES FROM NSAC AND DUKE POWER COMPANY.

WORDS RISK;PROBABILITY;OCONEE 3 (PWR);REACTOR, PWR;COMPARISON;EPRI

0/0000001-00007777 21

SSION NO. 00J0165265

LE PROCEEDINGS OF THE SECOND INTERNATIONAL SEMINAR ON STRUCTURAL RELIABILITY OF MECHANICAL COMPONENTS AND SUBASSEMBLIES OF NUCLEAR POWER PLANTS

HOR(S) BONILLA CF;JAEGER TA;FISTEDIS SH

1980

G

170 PPS. NUCLEAR ENGINEERING & DESIGN, 60(1), (SEPT. 1980). (MEETING HELD IN CONJUNCTION WITH THE 5TH SMIRT CONFERENCE. AUG. 1979)

GORY 010000;110000;230000

TION 0127

TRY G

NEDE

RACT EMPHASES WERE FOCUSED ON THE UTILIZATION AND ACCEPTANCE OF PROBABILISTIC STRUCTURAL RELIABILITY METHODOLOGY FOR CRITICAL NUCLEAR COMPONENTS IN THE FOUR FOLLOWING PROBLEM AREAS: THE ROLE OF PROBABILISTIC METHODS IN THE DEVELOPMENT OF REGULATIONS FOR NUCLEAR POWER PLANT DESIGN; UTILIZATION OF PRESENT KNOWLEDGE OF PROBABILISTIC STRUCTURAL RELIABILITY IN ANALYSES OF NUCLEAR POWER PLANTS; SYSTEMS RELIABILITY/STRUCTURAL RELIABILITY - INTERACTION AND DIFFERENCES IN APPROACH; AND FUTURE DEVELOPMENTS OF PROBABILISTIC STRUCTURAL RELIABILITY. GENERALLY SPEAKING A RATHER POSITIVE VIEW EVOLVED WITH RESPECT TO THE PRESENT UTILIZATION OF PROBABILISTIC METHODS IN THE ANALYSIS OF NUCLEAR POWER PLANTS. SEVERAL ITEMS, SUCH AS THE INFLUENCE OF HUMAN AND DESIGN ERRORS ETC., WERE NOT DISCUSSED. THE NEXT SEMINAR OF THIS KIND IS GOING TO BE HELD FOLLOWING THE 1981 MEETING OF SMIRT-6 IN PARIS.

ORDS GERMANY;PROBABILITY;STRUCTURAL INTEGRITY;STRUCTURAL ANALYSIS, DYNAMIC;RELIABILITY ANALYSIS;RISK;POWER PLANT, NUCLEAR

0/0000001-00007777 22

SSION NO. 00J0165129

THE IMPLICATIONS OF THE GERMAN RISK STUDY

R(S) BIRKHOFER A ;KOEBERLEIN K

UTH GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS), GARCHING, F.R. GERMANY

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        1980
E       U
        4 PPS. NUCLEAR ENGINEERING INTERNATIONAL, 23(306), PP. 53-55
        (NOV. 1980)
EGORY   010000;230000
TION    0126
P CODE  GRS
NTRY    U
BJ      NEIN
TRACT   THE METHODS AND RESULTS OF THE GERMAN RISK STUDY PUBLISHED A
        YEAR AGO ARE SUMMARIZED AND ITS IMPLICATIONS FOR REACTOR SAFETY
        ARE DISCUSSED.  IT HAS LED TO SUGGESTIONS THAT RISK ANALYSIS
        SHOULD BE MORE WIDELY USED FOR NUCLEAR AND OTHER TECHNOLOGICAL
        SYSTEMS.  IT HAS ALSO IDENTIFIED THE NEED FOR SPECIFIC SYSTEM
        MODIFICATIONS AND CONFIRMED TRENDS IN SAFETY RESEARCH.
WORDS   GERMANY;RISK,PROBABILITY;FAULT TREE ANALYSIS;CORE MELTDOWN;
        OPERATION;TRANSIENT;FISSION PRODUCT RELEASE;STEAM;EXPLOSION

/0/0000001-000077//   23
ESSION NO.  00X0164972
LE      STRUCTURAL UNCERTAINTY IN SEISMIC RISK ANALYSIS
HOR(S)  HASSELMAN TK;SIMONIAN SS
PAUTH   LAWRENCE LIVERMORE LAB., CALIF.
E       1980
E       N
D       NUREG/CR-1560 + UCRL-15216 +.  139 PPS, 5 TABS, 38 FIGS, OCT.
        1980
IL      AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S.
        DEPARTMENT OF COMMERCE, SPRINGFIELD, VA.  22161
EGORY   020000;230000;110000
TION    0126
P CODE  LZC
NTRY    A
TRACT   THIS REPORT DOCUMENTS THE FORMULATION OF A METHODOLOGY FOR
        MODELING AND EVALUATING THE EFFECTS OF STRUCTURAL UNCERTAINTY
        ON PREDICTED MODAL CHARACTERISTICS OF THE MAJOR STRUCTURES AND
        SUBSTRUCTURES OF COMMERCIAL NUCLEAR POWER PLANTS.  THE
        UNCERTAINTIES ARE CAST IN THE FORM OF NORMALIZED RANDOM
        VARIABLES WHICH REPRESENT THE DEMONSTRATED ABILITY TO PREDICT
        MODAL FREQUENCIES, DAMPING AND MODAL RESPONSE AMPLITUDES FOR
        BROAD GENERIC TYPES OF STRUCTURES (STEEL FRAME, REINFORCED
        CONCRETE AND PRESTRESSED CONCRETE).
WORDS   SITING;SEISMIC DESIGN;DATA PROCESSING;ANALYTICAL MODEL;HAZARDS
        ANALYSIS;STRUCTURAL ANALYSIS, DYNAMIC;RISK;PROBABILITY;
        MATHEMATICAL STUDY;CONCRETE;HJCK;NRC-RD;NRC-RM;INTERACTION,
        FOUNDATION AND STRUCTURE

/0/0000001-000077//   24
ESSION NO.  00X0164509
E       THE CHARACTERIZATION AND EVALUATION OF UNCERTAINTY IN
        PROBABILISTIC RISK ANALYSIS
OR(S)   PARRY GW;WINTER PW
AUTH    UKAEA SAFETY & RELIABILITY DIRECTORATE, WARRINGTON, U.K.
        1980
        N
        SRD R 190 +.  35 PPS, 6 FIGS, 114 REFS, OCT. 1980
L       AVAILABILITY - THE EDITOR, UNITED KINGDOM ATOMIC ENERGY
        AUTHORITY, SAFETY & RELIABILITY DIRECTORATE, CULCHETH,
        WARRINGTON WA3 4NE, ENGLAND
GORY    230000
ION     0125
 CODE   UKA
TRY     U
RACT    THE SOURCES OF UNCERTAINTY IN PROBABILISTIC RISK ANALYSIS ARE
        DISCUSSED USING THE EVENT/FAULT TREE METHODOLOGY AS AN EXAMPLE.
        THE ROLE OF STATISTICS IN QUANTIFYING THESE UNCERTAINTIES IS
        INVESTIGATED.  A CLASS OF UNCERTAINTIES IS IDENTIFIED WHICH IS,
        AT PRESENT, UNQUANTIFIABLE, USING EITHER CLASSICAL OR BAYESIAN
        STATISTICS.  IT IS ARGUED THAT BAYESIAN STATISTICS IS THE MORE
        APPROPRIATE VEHICLE FOR THE PROBABILISTIC ANALYSIS OF RARE
        EVENTS AND A SHORT REVIEW IS GIVEN WITH SOME DISCUSSION ON THE
        REPRESENTATION OF IGNORANCE.
RDS     RISK;BENEFIT VS RISK;PROBABILITY;FAULT TREE ANALYSIS;ERROR
        ANALYSIS
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SSION NO. 00C0164491
LE NATO ADVANCED STUDY INSTITUTE ON SYNTHESIS AND ANALYSIS METHODS FOR SAFETY AND RELIABILITY STUDIES
HOR(S) APOSTOLAKIS G;VOLTA G ;GARRIBBA S
PAUTH UNIV. OF CALIF., LOS ANGELES ;. C.E.C. JOINT RESEARCH CENTRE, ISPRA, ITALY
E 1978
PE L
D 463 PPS. BOOK PUBLISHED BY PLENUM PRESS, N.Y., 1980; PROCEEDINGS OF THE MEETING HELD AT URBINO, ITALY, JULY 3-14, 1978 (ISBN 0-306-40316-1)
EGORY 230000
TION 0125
P CODE UAV
NTRY I
TRACT THE INSTITUTE WAS COMPOSED OF LECTURES, WORKSHOPS AND GUIDED DISCUSSIONS. FROM THE LARGE QUANTITY OF WRITTEN MATERIAL THAT WAS USED AND PRODUCED DURING THE INSTITUTE, A NUMBER OF PAPERS INTRODUCING THE MOST RELEVANT RESEARCH RESULTS AND TRENDS IN THE FIELD HAVE BEEN SELECTED. THE PAPERS HAVE BEEN EDITED, PARTLY REWRITTEN AND REARRANGED IN ORDER TO OBTAIN IN THE END AN INTEGRATED EXPOSITION OF METHODS AND TECHNIQUES FOR RELIABILITY ANALYSIS AND COMPUTATION OF COMPLEX SYSTEMS.
WORDS SAFETY ANALYSIS;RISK;PROBABILITY;RELIABILITY ANALYSIS; RELIABILITY, SYSTEM;INTERNATIONAL;FAULT TREE ANALYSIS;FAILURE, COMMON MODE

0/0000001-000077//   26
SSION NO. 00J0163066
LE THE GERMAN REACTOR SAFETY STUDY
HOR(S) BIRKHOFER A
  1980
  G
  6 PPS, 6 FIGS. ATOMWIRTSCHAFT/ATOMTECHNIK, 25(10), PP. 515-20 (OCT. 1980)
GUAGE GERMAN
EGORY 010000;180000
TION 0123
NTRY G
E AWAK
RACT THE MOST IMPORTANT RESULTS OF THE GERMAN RISK STUDY OF A NUCLEAR POWER PLANT EQUIPPED WITH A PRESSURIZED WATER REACTOR WERE PUBLISHED IN AUGUST 1979. EIGHT TECHNICAL VOLUMES CONTAIN DETAILED DESCRIPTIONS AND DOCUMENTATIONS OF THE INVESTIGATIONS CARRIED OUT. THE REFERENCE FACILITY USED AS A BASIS FOR THE TECHNICAL PLANT STUDIES WERE UNIT B OF THE BIBLIS NUCLEAR POWER STATION, A KWU PRESSURIZED WATER REACTOR OF 3750 MW THERMAL POWER. THIS CONTRIBUTION PROVIDES MORE DETAILED EXPLANATIONS OF THE METHODS AND THE RESULTS OF THE RISK STUDY ILLUSTRATED BY EXAMPLES. (EWH)
ORDS GERMANY;SAFETY EVALUATION;ACCIDENT ANALYSIS;RISK;FORECAST; PROBABILITY;REACTOR, PWR;REVIEW;DESIGN STUDY;SAFETY ANALYSIS

0/0000001-000077//   27
SSION NO. 00C0163039
E VALUE/IMPACT COMPARISON OF ALTERNATE CONTAINMENT DESIGNS
OR(S) CARLSON DD;HICKMAN JW;TAYLOR MA
AUTH SANDIA LABS., ALBUQUERQUE, N.M.
  1977
  L
  SAND77-1103C + CONF-771109-43 +. 6 PPS, FROM ANS WINTER MEETING; SAN FRANCISCO, CALIF., NOV. 27, 1977
  AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GORY 110000;010000;230000
ION 0123
CODE AUA
RY A
ACT A VALUE/IMPACT ASSESSMENT IS MADE OF ALTERNATE CONTAINMENT CONCEPTS FOR COMMERCIAL LIGHT WATER REACTOR POWER PLANTS. SEVERAL ALTERNATE CONTAINMENT CONCEPTS ARE EVALUATED AND COMPARED, CONSIDERING THEIR POTENTIAL FOR REDUCING PUBLIC RISK AND THEIR CONSTRUCTION COST. THE RESULTS AND METHODOLOGY OF THE REACTOR SAFETY STUDY (WASH-1400) ARE USED AS A BASIS FOR

DETERMINING POTENTIAL RISK REDUCTIONS RESULTING FROM THE ALTERNATE CONTAINMENT DESIGNS. AMONG THE ALTERNATIVES CONSIDERED, FILTERED ATMOSPHERIC VENTING OFFERS THE GREATEST POTENTIAL FOR REDUCING PUBLIC RISK FOR THE LEAST IMPACT.

WORDS CONTAINMENT;CONTAINMENT, UNDERGROUND;CONTAINMENT, PRESSURE VENTING;PROBABILITY;RISK;REACTOR, BWR;REACTOR, PWR;COMPARTMENT; FAILURE MODE ANALYSIS;ACCIDENT, CORE DISRUPTIVE

3/07/0000001-00000777/ 28

CESSION NO. 00R0162863

TLE HTGR-PROCESS STEAM/COGENERATION AND HTGR-STEAM CYCLE PROGRAM. SEMIANNUAL REPORT FOR THE PERIOD OCTOBER 1, 1979 THROUGH MARCH 31, 1980

RPAUTH GENERAL ATOMIC CO., SAN DIEGO, CALIF.

TE 1980

PE G

10 GA-A16057 +. 90 PPS, TABS, FIGS, SEPT. 1980

IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

EGORY 120000

TION 0122

P CODE LAW

NTRY A

TRACT PROGRESS IN THE DESIGN OF AN 1170-MW(T) HIGH-TEMPERATURE GAS-COOLED REACTOR (HTGR) NUCLEAR STEAM SUPPLY (NSS) IS DESCRIBED. THIS NSS IS SHOWN TO INTEGRATE FAVORABLY INTO PRESENT PETROCHEMICAL AND PRIMARY METAL PROCESS INDUSTRIES, INTO HEAVY OIL RECOVERY OPERATIONS, AND INTO FUTURE SHALE OIL RECOVERY AND SYNFUEL PROCESSES. COST ESTIMATES FOR CENTRAL STATION POWER-GENERATING 2240- AND 3360-MW(T) HTGR-STEAM CYCLE (HTGR-SC) PLANTS ARE UPDATED. THE 2240-MW(T) HTGR-SC IS TREATED TO A PROBABILISTIC RISK EVALUATION.

WORDS REACTOR, HTGR;ECONOMICS;COST ANALYSIS;RISK;PROBABILITY;STEAM; FUEL, FOSSIL

/07/0000001-00000777/ 29

ESSION NO. 00X0162863

LE A USER'S GUIDE FOR MODCUT AND PL-MODMC: COMPUTER CODES FOR FAULT TREE ANALYSIS

HOR(S) MODARRES M;RASMUSSEN NC;WOLF L

PAUTH MASS. INST. OF TECHNOLOGY, CAMBRIDGE

E 1980

E N

D NUREG/CR-1461 +. 49 PPS, 8 REFS, NOV. 1980

IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

EGORY 090000

TION 0122

P CODE MOM

TRY A

RACT THESE COMPUTER PROGRAMS WERE DEVELOPED TO AID IN GENERATING CUTSETS AND PERFORMING MONTE-CARLO ANALYSIS OF COMPLEX FAULT TREES. THE MODCUT ALGORITHM IS BASED ON DERIVING ALL SIMPLE CUTSETS FROM THE MODULAR CUTSETS. FIRST, SIMPLE CUTSETS OF ALL MODULES IN THE MODULAR CUTSETS ARE CALCULATED. THEN BY PROPERLY ASSIGNING THESE CUTSETS TO THE CORRESPONDING MODULAR CUTSETS, A COMPLETE SET OF SIMPLE CUTSETS FOR THE TOP EVENT OF THE TREE ARE DETERMINED. THE CODE PL-MODMC WAS DEVLOPED TO INCORPORATE UNCERTAINTY ANALYSIS CAPABILITY INTO PL-MOD USING MONTE-CARLO SIMULATION APPLIED TO THE MODULAR CUTSETS THAT PL-MOD GENERATES.

ORDS COMPUTER PROGRAM;FAULT TREE ANALYSIS;MONTE CARLO;RISK; PROBABILITY;HJCK;NRC-4

/07/0000001-00000777/ 30

SSION NO. 00J0159831

E APPLICATION OF SIMPLIFIED RELIABILITY METHODS FOR RISK ASSESSMENT OF NUCLEAR WASTE REPOSITORIES

OR(S) GASSMANN J;PRITZKER A

AUTH MOTOR-COLUMBUS CONSULTING ENGINEERS INC., BADEN, SWITZERLAND

1980

D

8 PPS, 3 TABS, 6 FIGS, 5 REFS. NUCLEAR TECHNOLOGY, 48(3), PP. 289-97 (MAY 1980)

EGORY 140000;230000
TION 0117
NTRY A
BS NUAT
TRACT THIS PAPER PRESENTS A PROBABILISTIC METHOD FOR THE ESTIMATION OF THE RISK OF RADIOACTIVE RELEASES FROM UNDERGROUND WASTE REPOSITORIES. THE METHOD IS BASED ON SIMPLE PROBABILISTIC MODELS AND CONTAINS ONLY A FEW PARAMETERS. ITS APPLICATION IS FAST AND ALLOWS A PRELIMINARY ASSESSMENT OF REPOSITORY CONCEPTS WITH NATURAL AND MAN-MADE BARRIERS IN THE EARLY SITE EVALUATION AND DESIGN PHASE. (EWH)
YWORDS RELIABILITY ANALYSIS;RISK;WASTE STORAGE;WASTE DISPOSAL, TERRESTRIAL;WASTE DISPOSAL, BEDROCK;ANALYTICAL MODEL; PROBABILITY;RADIOACTIVITY RELEASE

/0/0000001-0000077// 31
ESSION NO. 00C0159744
LE OYSTER CREEK PROBABILISTIC SAFETY ANALYSIS (OPSA)
HOR(S) GARRICK BJ;KAPLAN S
PAUTH PICKARD, LOWE & GARRICK INC., IRVINE, CALIF. 92715
E 1980
E L
O 8 PPS. FROM 1980 ANS/ENS TOPICAL MEETING ON THERMAL REACTOR SAFETY; KNOXVILLE, TENN., APRIL 7-11, 1980
IL AVAILABILITY - B.J. GARRICK, PICKARD, LOWE & GARRICK INC., IRVINE, CALIF. 92715
EGORY 170000;010000;060000;050000;020000;120000
TION 0117
NTRY A
TRACT HIGHLIGHTS ARE PRESENTED OF AN INDEPENDENT PROBABILISTIC RISK ASSESSMENT OF THE OYSTER CREEK NUCLEAR PLANT. ELEMENTS OF STUDY INCLUDED RELEASE FREQUENCIES, COMMON CAUSE ANALYSIS, AND CONSEQUENCE ANALYSIS. WHILE STUDY WAS BASED ON WASH-1400 METHODOLOGY, ADVANCES IN RISK ANALYSIS WERE USED IN QUANTIFYING UNCERTAINTY AND MODELING SITE SPECIFIC CHARACTERISTICS. STUDY PROVIDES A BASIS FOR EVALUATING IMPACT ON RISK OF PLANT MODIFICATIONS AND PROCEDURAL CHANGES.
WORDS REACTOR, BWR;OYSTER CREEK (BWR);FAULT TREE ANALYSIS;RISK; PROBABILITY;SAFETY ANALYSIS;CORE MELTDOWN;SEISMIC DESIGN; ACCIDENT, CONSEQUENCES;ANALYTICAL MODEL;PLUME BEHAVIOR;FAILURE, SEQUENTIAL

/0/0000001-0000077// 32
ESSION NO. 00C0158989
LE EVALUATION OF THE THREE MILE ISLAND ACCIDENT IN THE CONTEXT OF WASH-1400
HOR(S) BURNS RD
PAUTH LOS ALAMOS SCIENTIFIC LAB., N.M.
1980
L
5 PPS. FROM 1980 ANS/ENS TOPICAL MEETING ON THERMAL REACTOR SAFETY; KNOXVILLE, TENN., APRIL 7-11, 1980
L AVAILABILITY - ROBERT D. BURNS III, LOS ALAMOS SCIENTIFIC LAB., LOS ALAMOS, N.M. 87545
GORY 170000;010000
TION 0115
CODE AOA
TRY A
RACT COMPARISON OF WASH-1400 REACTOR SAFETY STUDY WITH COMMERCIAL REACTOR EXPERIENCE SHOWS THAT TMI ACCIDENT DOES NOT CHALLENGE VALIDITY OF WASH-1400. SEVERITY OF ACCIDENT WAS CONSISTENT WITH A "PWR-8" CATEGORY ACCIDENT AS DESCRIBED IN STUDY. EXACT SEQUENCE OF FAILURES IN TMI ACCIDENT IS NOT INCLUDED IN WASH-1400, BECAUSE OF DESIGN DIFFERENCES BETWEEN REFERENCE WESTINGHOUSE PWR USED FOR STUDY AND BABCOCK AND WILCOX PWR AT TMI. HOWEVER, TMI SEQUENCE IS INCLUDED IN WASH-1400 GENERAL DESCRIPTION OF TRANSIENT-INITIATED ACCIDENTS. PROBABILITY ANALYSIS SHOWS THAT OCCURRENCE OF TMI ACCIDENT IS CONSISTENT WITH WASH-1400 PROBABILITY ESTIMATES.
ORDS REACTOR, PWR;THREE MILE ISLAND 2 (PWR);INCIDENT;PROBABILITY; RISK;RADIATION EFFECT, COMMUNITY;FAILURE, SEQUENTIAL

SSION NO.   00E0158552
LE   RISK ASSESSMENT OF MAJOR FIRES IN AN HTGR PLANT
HOR(S)   FLEMING KN
PAUTH   GENERAL ATOMIC CO., SAN DIEGO, CALIF.
E   1980
E   H
N   GA-A15622 +. 22 PPS, TABS, FIGS, APRIL 1980
IL   AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY   170000;230000;180000
TION   0114
P CODE   LAW
NTRY   A
TRACT   HTGR RISK ASSESSMENT STUDY HAS BEEN EXTENDED TO INCLUDE MAJOR FIRES AS INITIATING EVENTS. MAJOR ASPECTS OF STUDY INCLUDE DEVELOPMENT OF METHODOLOGY, COLLECTION AND INTERPRETATION OF FIRE EXPERIENCE DATA AND APPLICATION OF METHODS AND DATA TO AN HTGR PLANT. QUALITATIVE AND QUANTITATIVE METHODS WERE DERIVED TO IDENTIFY IMPORTANT FIRE LOCATIONS. FIRE PROPAGATION MODEL WAS USED IN CONJUNCTION WITH EXPERIENCE DATA AND DETAILED FAULT TREE ANALYSES TO ESTIMATE COMMON CAUSE FAILURE PROBABILITIES ASSOCIATED WITH A SPECTRUM OF POTENTIAL FIRES. IT WAS DETERMINED THAT FIRES MAKE A SIGNIFICANT CONTRIBUTION TO THE HTGR RISK ASSESSMENT ONLY AT ACCIDENT FREQUENCY LEVELS BELOW 10(-7)/REACTOR-YEAR.
WORDS   REACTOR, HTGR;FIRE;RISK;INCIDENT COMPILATION;FAULT TREE ANALYSIS;PROBABILITY;OPERATING EXPERIENCE SUMMARY;POWER PLANT, NUCLEAR

/0/0000001-0000077//   34
ESSION NO.   0020158216
LE   AN EVALUATION OF THE RESIDUAL RISK FROM THE INDIAN POINT NUCLEAR POWER PLANTS
PAUTH   OFFSHORE POWER SYSTEMS, JACKSONVILLE, FLA.
E   1980
E   G
N   APPROX. 85 PPS, LTR W/REPORT NO. 35A96 TO U.S. NRC, MAY 23, 1980 (DOCKET 50-247/286)
IL   AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)
EGORY   170000;010000;120000;090000;110000;230000
TION   0113
P CODE   OPS
NTRY   A
TRACT   REPORT IS IN RESPONCE TO NRC PRESENTATION TO ACRS WHICH STATED THAT ZION AND INDIAN PT. PLANTS COMPRISE MORE THAN 30 PERCENT OF NATIONAL RISK FROM NUCLEAR REACTORS. REPORT PRESENTS ESTIMATE OF RESIDUAL RISK ASSOCIATED WITH INDIAN POINT UNITS 2 AND 3. STUDY WAS BASED ON METHODOLOGY AND DATA FOR WASH 1400. CONCLUSION WAS THAT LEVEL OF RISK ASSOCIATED WITH INDIAN POINT PLANTS IS SIGNIFICANTLY LESS THAN LEVEL OF RISK WHICH HAS FOUND IMPLICIT ACCEPTANCE IN PAST NRC LICENSING ACTIONS, THE LEVEL OF RISK REPORTED IN WASH 1400 FOR A TYPICAL PWR LOCATED AT AN AVERAGE OR COMPOSITE SITE. CONCLUSION RESULTS FROM APPLICATION OF WASH 1400 METHODOLOGY TO SPECIFIC DESIGN OF INDIAN POINT PLANTS AND SITE SPECIFIC CHARACTERISTICS SUCH AS DEMOGRAPHY, METEOROLOGY, ETC.
ORDS   REACTOR, PWR;INDIAN POINT 2 (PWR);INDIAN POINT 3 (PWR);AGENCY, NRC;ACRS;RISK;RELIABILITY ANALYSIS;PROBABILITY;ACCIDENT; ACCIDENT, CONSEQUENCES;MAIN COOLING SYSTEM;SYSTEM ANALYSIS; RELIABILITY, SYSTEM;CONTAINMENT SYSTEM, OPERATION

0/0000001-0000077//   35
SSION NO.   0030158184
E   PROBABILISTIC RISK STUDIES EARN INCREASED ACCEPTANCE WITH REGULATORS
    1980
    G
    2 PPS, NUCLEAR INDUSTRY, 27(4), PP. 14-15 (APRIL 1980)
GORY   010000
ION   0113
TRY   A
    NUID

RACT   THE ACCIDENT AT THREE MILE ISLAND MADE BELIEVERS OUT OF SOME IN THE INDUSTRY, AND THE NRC, WHO HAD BEEN SKEPTICAL OF THE VALUE OF PROBABILISTIC RISK ASSESSMENT AS AN EFFECTIVE TOOL IN EVALUATING THE SAFETY OF REACTOR SYSTEMS. ONE YEAR AGO, NOT ONE NRC STAFF MEMBER COULD STATE THAT THE RASMUSSEN RISK STUDY HAD BEEN A FACTOR IN ANY LICENSING DECISION. NOW IT IS BEING EMBRACED AS THE METHOD FOR QUANTIFYING SAFETY IN A MANNER THAT WILL BE ACCEPTABLE TO THE PUBLIC.
WORDS   RISK;BENEFIT VS RISK;PROBABILITY;ACCIDENT, PROBABILITY OF;THREE MILE ISLAND 2 (PWR);INCIDENT;REACTOR, PWR

0/0000001-00007777   36
ESSION NO.   00R0157939
LE   RISK ANALYSIS METHODS DEVELOPMENT ELEVENTH QUARTERLY REPORT JULY-SEPTEMBER 1979
PAUTH   GENERAL ELECTRIC CO., SUNNYVALE, CALIF.
E   1979
E   G
D   GEFR-14023-11 +. 13 PPS, 5 FIGS, 8 REFS, OCT. 1979
IL   AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)
EGORY   010000
TION   0113
P CODE   GEC
NTRY   A
TRACT   THE OBJECTIVES OF THIS PROGRAM ARE: 1. TO DEVELOP THE TECHNICAL BASIS FOR PERFORMANCE OF CREDIBLE BREEDER REACTOR (BR) RISK ASSESSMENTS. 2. TO DEVELOP AND APPLY METHODS FOR R&D PLANNING IN SUPPORT OF THE LINE OF ASSURANCE (LOA) STRATEGY. THIS QUARTERLY REPORT DESCRIBES ANALYSES DONE UNDER SUBTASK C, LOA SUPPORTING ANALYSIS, AND SUBTASK E, METHODS AND PROCEDURES DEVELOPMENT. THE WORK ON SUBTASK C DESCRIBES THE PARAMETRIC ANALYSIS PERFORMED TO DETERMINE THE EFFECT OF HODOSCOPE SPACING AND SODIUM VOIDING LOCATION ASSUMPTIONS ON THE SUCCESS PREDICTIONS OBTAINED FROM THE LOA-2/TOP GEDANKEN EXPERIMENT. SUBTASK E WORK IS AN EXAMPLE APPLICATION OF THE SINGLE PLANT RISK MODEL. (FAH)
ORDS   RISK;ACCIDENT ANALYSIS;REACTOR, BREEDER;R AND D PROGRAM; PROBABILITY;ANALYTICAL MODEL

0/0000001-00007777   37
SSION NO.   0020157936
E   IMPLEMENTATION OF PROBABILISTIC RISK TECHNIQUES (PRT)
AUTH   BABCOCK & WILCOX, LYNCHBURG, VA
    1980
    G
    2 PGS, LTR TO NRC DIRECTOR OF OFFICE OF NUCLEAR REACTOR REGULATION, MAY 14, 1980
L   AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)
GORY   010000;170000;180000
ION   0113
CODE   MAL
RY   A
RACT   LETTER DISCUSSES B&W'S PERSPECTIVE REGARDING IMPLEMENTATION OF PRT AND REQUESTS CLARIFICATION OF NRC'S PROPOSED COURSE OF ACTION. B&W BELIEVES THAT NRC SHOULD AWAIT RESULTS OF STUDY UNDERWAY BY VARIOUS GROUPS BEFORE IMPLEMENTING RISK-BASED CRITERIA FOR DESIGN AND OPERATION OF NUCLEAR POWER PLANTS. BELIEVES THAT PREMATURE IMPOSITION OF RISK-BASED REGULATIONS WILL DISCREDIT LONG-TERM EFFORTS UNDERWAY, ENCOURAGE PUBLIC MISCONCEPTION, PLACE UNFAIR BURDEN ON UTILITIES AND RATE PAYERS, AND BE DETRIMENTAL TO ADVANCING STATE OF ART IN NUCLEAR PLANT DESIGN, OPERATION AND REGULATION.
RDS   REACTOR, POWER;AGENCY, NRC;REGULATION, NRC;RISK;RELIABILITY ANALYSIS;DESIGN STUDY;PROBABILITY

/0000001-00007777   38
SION NO.   0030157870
    THE UNCERTAINTY IN ACCIDENT CONSEQUENCES CALCULATED BY LARGE CODES DUE TO UNCERTAINTIES IN INPUT
R(S)   NGUYEN DH

AUTH HANFORD ENGINEERING DEVELOPMENT LAB., RICHLAND, WASH.
1980
D
12 PPS, 6 FIGS. 25 REFS, NUCLEAR TECHNOLOGY, 49(1), PP. 80-91 (JUNE 1980)
EGORY 050000
TION 0113
P CODE WAD
NTRY A
B NUAT
TRACT A SENSITIVITY ANALYSIS WAS FIRST APPLIED TO THE CODE TO SCREEN THE INPUT VARIABLES, LEAVING ONLY THOSE MOST AFFECTING THE OUTPUT CONSEQUENCES. THE VARIATIONS OF THESE EFFECTIVE INPUTS WERE PRESCRIBED BY AN EFFECTIVE COMBINATION OF STATISTICAL DESIGNS, WHICH ACCOUNTED FOR THE LINEAR, QUADRATIC, AND TWO-FACTOR INTERACTION EFFECTS OF THE INPUTS ON THE CALCULATED CONSEQUENCE. A KEY RESULT OF THE METHODOLOGY WAS THE PROBABILITY DENSITY FUNCTION OF THE CONSEQUENCE OF INTEREST, EXPRESSED AS A DISTRIBUTION OF THE PEARSON FAMILY. THE CONFIDENCE LEVEL IN CALCULATING A CONSEQUENCE WAS READILY OBTAINED FROM THIS DISTRIBUTION FUNCTION. THE METHODOLOGY WAS APPLIED TO THE COMPUTER CODE MELT-IIIA AND THE CONFIDENCE LEVEL IN PREDICTING THE TIME OF INITIAL PIN FAILURE DURING A TRANSIENT OVERPOWER ACCIDENT IN THE FAST TEST REACTOR WAS DETERMINED.
WORDS COMPUTER PROGRAM;RISK;PROBABILITY;SENSITIVITY ANALYSIS;REACTOR, LMFBR;ACCIDENT, CORE DISRUPTIVE

0/0000001-00000777/ 39
SSION NO. 00V0157097
E NRC STAFF REQUEST EVALUATION REVIEW OF LIMERICK NUCLEAR STATION
AUTH U.S. NUCLEAR REGULATORY COMMISSION
1980
M
NRC NEWS RELEASE 80-87 +. 1 PG, FOR WEEK ENDING MAY 13, 1980
L AVAILABILITY - NRC, OFFICE OF PUBLIC AFFAIRS, WASHINGTON, D.C. 20555
GORY 180000;020000
ION 0111
CODE NRC
TRY A
RACT NRC'S OFFICE OF NUCLEAR REACTOR REGULATION HAS REQUESTED THAT PHILADELPHIA ELECTRIC COMPANY CONDUCT A RISK ASSESSMENT REVIEW OF ITS TWO-UNIT LIMERICK NUCLEAR GENERATING STATION IN VIEW OF THE HIGH POPULATION DENSITY SURROUNDING THE SITE. IN CALLING FOR THE RISK ASSESSMENT STUDY, THE NRC STAFF REQUESTED THAT PHILADELPHIA ELECTRIC DETERMINE THE DOMINANT CONTRIBUTORS TO RISK, THROUGH PROBABILISTIC TECHNIQUES, AND IF THE LIMERICK SAFETY FEATURES COMPENSATE FOR THE HIGH POPULATION DENSITY. THIS STUDY WILL UTILIZE SOME OF THE TECHNICES USED IN THE DEVELOPMENT OF THE NRC'S REACTOR SAFETY STUDY, WASH-1400.
ORDS AGENCY, NRC;REVIEW;LIMERICK 1 (BWR);LIMERICK 2 (BWR);REACTOR, BWR;RISK;PROBABILITY;POPULATION;POPULATION DISTRIBUTION; METEOROLOGY;HYDROLOGY;SITING, REACTOR

0/0000001-00000777/ 40
SSION NO. 00X0156749
E GUIDELINE FOR AUTOMATIC DATA PROCESSING RISK ANALYSIS
AUTH NATIONAL BUREAU OF STANDARDS
1979
M
FIPS-PUB-65 +. 27 PPS, 8 FIGS, 11 REFS, AUG. 1, 1979
L AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GORY 010000
ION 0111
TRY A
RACT THIS GUIDELINE EXPLAINS THE REASONS FOR PERFORMING A RISK ANALYSIS, DETAILS THE MANAGEMENT INVOLVEMENT NECESSARY AND PRESENTS PROCEDURES AND FORMS TO BE USED FOR RISK ANALYSIS AND COST EFFECTIVE EVALUATION OF SAFEGUARDS. RESEARCH IN THIS AREA WILL NOT BE DETERRED BY THE INFLEXIBILITY OF AN ALREADY PRESCRIBED METHODOLOGY BUT SHOULD BE ENCOURAGED BY THE SETTING OF THESE BASIC CRITERIA AND THE CHALLENGE OF DEVELOPING AND

REFINING MORE SOPHISTICATED AND MORE EASILY APPLIED TECHNIQUES.
WORDS COMPUTER PROGRAM;RISK;MATHEMATICAL TREATMENT;PROBABILITY; HAZARDS ANALYSIS

/07000001-00000777/ 41
ESSION NO. 00C0155706
TLE COMPARISON OF RISK OF VARIOUS ELECTRICAL ENERGY SOURCES
THOR(S) INHABER H
RPAUTH ATOMIC ENERGY CONTROL BOARD, OTTAWA, CANADA
TE 1980
PE L
RD 7 PPS. FROM 1980 ANS/ENS TOPICAL MEETING ON THERMAL REACTOR SAFETY; KNOXVILLE, TENN., APRIL 7-11, 1980
AIL AVAILABILITY - H. INHABER, ATOMIC ENERGY CONTROL BOARD, OTTAWA, CANADA K1P 5S9
TEGORY 010000
ITION 0109
UNTRY A
STRACT THE RISK TO HUMAN HEALTH OF VARIOUS ENERGY SOURCES HAS PROVOKED SOME CONTROVERSY IN THE PAST FEW YEARS, DUE TO INADEQUATE AND CONFLICTING DATA, QUESTIONS SUCH AS WHERE ONE DRAWS THE BOUNDARY LINE FOR A COMPLETE ENERGY SYSTEM AND HOW LOW PROBABILITY, HIGH-CONSEQUENCE ACCIDENTS AFFECTING THE PUBLIC CAN BE FACTORED IN. IN THIS PAPER, THE RESULTS OF TWO RECENT RELATIVE STUDIES ARE COMPARED. THE VALUES USED IN ESTIMATING NUCLEAR RISK IN VARIOUS LITERATURE REVIEWS ARE ALSO EVALUATED IN SOME DETAIL.
WORDS RISK;HAZARD, RELATIVE;HAZARDS ANALYSIS;ENERGY SOURCE;ELECTRIC POWER, ALTERNATE;COAL;SOLAR;N-POWER, SAFETY OF;PROBABILITY

/07000001-00000777/ 42
ESSION NO. 00C0155701
LE RISK ANALYSIS: TOWARD A STANDARD METHOD
HOR(S) SMITH KR
PAUTH RESOURCE SYSTEMS INST., HONOLULU, HAWAII
E 1980
E L
D 12 PPS. FROM 1980 ANS/ENS TOPICAL MEETING ON THERMAL REACTOR; KNOXVILLE, TENN., APRIL 7-11, 1980
IL AVAILABILITY - KIRK R. SMITH, RESOURCE SYSTEMS INST., EAST-WEST CENTER, HONOLULU, HAWAII 96848
EGORY 010000
TION 0109
NTRY A
TRACT THE RISKS OF DIFFERENT SYSTEMS OFTEN ARE NOT ONLY COMPARED INCONSISTENTLY, BUT ARE CALCULATED IN A MANNER INAPPROPRIATE TO THE DECISIONS BEING MADE ABOUT ENERGY TECHNOLOGIES. WITHOUT A CAREFULLY CONSTRUCTED AND AGREED-UPON FRAMEWORK FOR TABULATING RISKS, IT IS POSSIBLE TO COME TO NEARLY ANY CONCLUSION ABOUT COMPARATIVE HAZARDS. HOWEVER, ADHERENCE TO A FEW STRAIGHTFORWARD RULES (SOME BORROWED FROM FINANCIAL ACCOUNTING) IS SUFFICIENT TO ENSURE CONSISTENCY.
WORDS STANDARDIZATION;ANALYTICAL TECHNIQUE;MATHEMATICAL TREATMENT; PROBABILITY;RISK;RELIABILITY ANALYSIS

/07000001-00000777/ 43
ESSION NO. 00X0155700
LE A RISK METHODOLOGY PRESENTATION
HOR(S) ERDMANN RC;KELLY JE;RUMBLE ET
PAUTH SCIENCE APPLICATIONS INC., PALO ALTO, CALIF.
1979
N
EPRI-NP-79-1-LD +. 364 PPS. FIGS. JAN. 1979
L AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX10090, PALO ALTO, CALIF. 94303
GORY 090000;010000
TION 0109
CODE SAI
TRY A
RACT THIS REPORT DOCUMENTS THE MATERIAL PRESENTED TO UTILITY AND EPRI PARTICIPANTS DURING A ONE-DAY TUTORIAL ON RISK METHODOLOGY APPLIED TO NUCLEAR POWER SYSTEMS. BASIC PROBABILISTIC AND RISK CONCEPTS AND EXAMPLES OF HISTORICAL DATA WERE REVIEWED FOLLOWED BY A DESCRIPTION OF EVENT TREE AND FAULT TREE ANALYSIS

METHODOLOGIES AND ASSOCIATED EPRI COMPUTER CODES. RESULTS OF APPLYING THESE METHODS TO THE POTENTIAL PROBLEM AREAS OF SEISMIC RESPONSE AND ANTICIPATED TRANSIENTS WITHOUT SCRAM IN LIGHT WATER REACTOR SAFETY ARE ALSO PRESENTED. (EWH)
WORDS RISK;PROBABILITY;FAULT TREE ANALYSIS;ANALYTICAL TECHNIQUE;EPRI; SEISMIC DESIGN;ACCIDENT, ATWS

/0700000001-000007777   44
ESSION NO. 00C0155133
LE NUCLEAR REACTOR SAFETY SESSION: THERMAL REACTOR SAFETY-6: RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
E 1979
E L
U 11 PPS, PP. 476-88 OF TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY, VOL. 32, FROM 1979 ANNUAL MEETING; ATLANTA, GA., JUNE 3-7, 1979
EGORY 010000;220000;120000;060000
TION 0108
NTRY A
TRACT PAPERS PRESENTED INCLUDE: 1) THE CONSEQUENCE MODEL OF THE GERMAN REACTOR SAFETY STUDY, 2) FREQUENCY OF ANTICIPATED TRANSIENTS, 3) SODIUM COMPONENT RELIABILITY DATA COLLECTION AT CREDO, 4) PIPE RUPTURE PROBABILITY AS APPLIED TO PWR STEAM LINE BREAK, 5) QUANTITATIVE SYSTEM INTERACTIONS METHODOLOGY: APPLICATION TO PWR POWER SUPPLIES, 6) SABOTAGE RISK MODELING FOR LWRS, 7) BAYESIAN PREDICTION MODEL FOR FIRE OCCURRENCES IN NUCLEAR POWER PLANTS, AND 8) A STUDY OF NUCLEAR POWER PLANT FIRES.
WORDS REACTOR, THERMAL;SAFETY ANALYSIS;RISK;PROBABILITY;RELIABILITY ANALYSIS;ACCIDENT, CONSEQUENCES;GERMANY;REACTOR, FAST;DATA PROCESSING;ACCIDENT, STEAM LINE RUPTURE;SABOTAGE;FIRE; PREDICTION

/0700000001-000007777   45
ESSION NO. 00C0155131
LE NUCLEAR REACTOR SAFETY SESSION: THERMAL REACTOR SAFETY-6: APPLICATION OF RISK ASSESSMENT: TECHNIQUES IN OTHER COUNTRIES
E 1979
E L
6 PPS, PP. 462-67 OF TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY, VOL. 32, FROM 1979 ANNUAL MEETING; ATLANTA, GA., JUNE 3-7, 1979
GORY 010000
TION 0108
NTRY A
TRACT PAPERS PRESENTED INCLUDE: 1) COMMUNICATION TECHNIQUES FOR RISK ESTIMATES, 2) APPLICATION OF PROBABILISTIC RISK ASSESSMENT TECHNIQUES IN JAPAN, 3) RISK ASSESSMENT FOR TWO REACTOR SITES IN NORWAY, 4) A REACTOR SAFETY STUDY APPLIED TO THE FORSMARK 3 NUCLEAR POWER PLANT, AND 5) ASSESSMENT OF ACCIDENT RISKS IN GERMAN NUCLEAR POWER PLANTS.
WORDS REACTOR, THERMAL;SAFETY ANALYSIS;RISK;PROBABILITY;INTERNATIONAL; GERMANY;JAPAN;NORWAY;SWEDEN

0700000001-000007777   46
SSION NO. 00X0154647
E THE GERMAN RISK STUDY: SUMMARY (IN ENGLISH)
AUTH THE FEDERAL MINISTER OF RESEARCH AND TECHNOLOGY, F.R. GERMANY
1979
N
GERRSR-450 +. 49 PPS, 4 TABS, 12 FIGS, AUG. 15, 1979
L AVAILABILITY - CONTACT DR. G.L. BENNETT, U.S. NUCLEAR REGULATORY COMMISSION, OFFICE OF NUCLEAR REGULATORY RESEARCH, WASHINGTON, D.C.,F OR DISTRIBUTION INFORMATION.
GORY 010000
ION 0107
TRY G
RACT THE PRESENT REPORT CONSTITUTES ONLY THE FIRST PART OF THE GERMAN RISK STUDY (PHASE A). THE RESULTS OF PHASE A OF THE GERMAN RISK STUDY HAVE BEEN COMPILED IN A MAIN VOLUME CONTAINING A TOTAL OF 9 CHAPTERS (1. OBJECTIVES, LAYOUT AND ORGANIZATION OF THE STUDY; 2. FUNDAMENTAL REMARKS ON THE IDENTIFICATION OF RISKS; 3. THE NUCLEAR POWER PLANT; 4. SUBJECT MATTER AND METHOD OF THE RISK ANALYSIS; 5. RESULTS OF

THE EVENT TREE ANALYSIS; 6. RELEASE OF FISSION PRODUCTS; 7. ACCIDENT CONSEQUENCE MODEL; 8. RESULTS AND INHERENT SIGNIFICANCE OF THE RESULTS; 9. CONCLUSIONS). (EWH)
WORDS GERMANY;SAFETY EVALUATION;DESIGN STUDY;ACCIDENT ANALYSIS; PROBABILITY;RISK;POWER PLANT, NUCLEAR;REACTOR, LWR;POPULATION EXPOSURE;N-POWER, SAFETY OF;FOREIGN EXCHANGE

07000001-00007777 47
ESSION NO. 00C0153297
LE THE ROLE OF RISK ASSESSMENT IN THE NUCLEAR REGULATORY PROCESS
HOR(S) LEVINE S
PAUTH U.S. NUCLEAR REGULATORY COMMISSION
E 1979
E L
O 9 PPS, 2 TABS, 2 FIGS, ANNALS OF NUCLEAR ENERGY, 6(5), PP. 281-89 (1979)
EGORY 180000;010000
TION 0104
P CODE NRC
NTRY U
DS ANSE
TRACT RISK ASSESSMENT TECHNIQUES, BENEFITS AND LIMITATIONS, ARE DISCUSSED WITH WASH-1400 IN MIND. THE METHODOLOGY IS REVIEWED, RESERVATIONS ARE DISCUSSED, THE NEED FOR ESTABLISHING CRITERIA FOR ACCEPTABLE LEVELS OF RISK IS CITED, ITS APPLICABILITY TO ALTERNATIVE CONTAINMENT SYSTEMS IS MEASURED, AND THE APPLICATION OF RISK ASSESSMENT TO PROGRAM DEVELOPMENT IN GENERAL IS ANALYZED.
WORDS ACCIDENT, PROBABILITY OF;RISK;AGENCY, NRC;RELIABILITY ANALYSIS; BENEFIT VS RISK;SAFETY EVALUATION;SAFETY PROGRAM;PROBABILITY; REACTOR, LWR

07000001-00007777 48
ESSION NO. 00C0153261
LE APPLICATION OF PROBABILISTIC TECHNIQUES TO SEISMIC RISK ANALYSIS OF THE DIABLO CANYON PLANT
HOR(S) BRUNDT WK;FRAY RR;MOULIA TA
PAUTH PACIFIC GAS & ELECTRIC CO., SAN FRANCISCO, CALIF.
E 1978
E L
O CONF-780507-P3 +. 10 PPS, PP. XIV.5-1 THRU -10, FROM VOL. 3, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
L AVAILABILITY - T.A. MOULIA, PACIFIC GAS & ELECTRIC CO., DEPT. OF MECHANICAL & NUCLEAR ENGINEERING, 77 BEALE ST., SAN FRANCISCO, CALIF. 94127
GORY 020000;180000
TION 0104
CODE PGE
TRY A
TRACT THE PRINCIPAL PURPOSE OF THIS STUDY WAS TO MAKE AN ASSESSMENT OF POTENTIAL RISKS TO THE PUBLIC FROM EARTHQUAKE-CAUSED ACCIDENTS AT THE DIABLO CANYON NUCLEAR POWER PLANT. IT WAS FOUND THAT EVEN WHEN CONSIDERING THE IMPACT ON THE PLANT OF ACCELERATIONS WELL IN EXCESS OF THE MAXIMUM WHICH THE NRC HAS ASKED THE COMPANY TO CONSIDER, THE RISK TO THE HEALTH AND SAFETY OF THE PUBLIC CAUSED BY POSSIBLE EARTHQUAKE DAMAGE TO THE PLANT WAS FOUND TO BE EXTREMELY REMOTE. (EWH)
ORDS PROBABILITY;SEISMIC DESIGN;RISK;DIABLO CANYON 1 (PWR);DIABLO CANYON 2 (PWR);SEISMOLOGY;EARTHQUAKE

07000001-00007777 49
SSION NO. 00C0153258
E AN EVALUATION OF THE INCREMENTAL SEISMIC RISK DUE TO THE PRESENCE OF NUCLEAR POWER PLANTS
OR(S) LEE YT;ORRENT D;APOSTOLAKIS G
AUTH UNIV. OF CALIF., LOS ANGELES
1978
L
CONF-780507-P3 +. 12 PPS, PP. XIV.2-1 THRU -12, FROM VOL. 3, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978

L AVAILABILITY - YUM TONG LEE, CHEMICAL, NUCLEAR, AND THERMAL ENGINEERING DEPT., SCHOOL OF ENGINEERING AND APPLIED SCIENCE. UNIV. OF CALIF., LOS ANGELES, CALIF. 90024
GORY 010000;110000
TION 0104
CODE UAV
NTRY A
TRACT THE SEISMIC RISK FOR THE CONTINENTAL UNITED STATES, IN TERMS OF THE EXPECTED ANNUAL NUMBER OF DEATHS AND SEVERE INJURIES, AND THE EXPECTED PROPERTY DAMAGE, IS EVALUATED IN THIS WORK. PROBABILISTIC MODELS AND CORRELATIONS ARE DEVELOPED AND USED IN THE EVALUATIONS OF THE RISKS, ACCOUNTING FOR SUCH IMPORTANT VARIABLES AS THE VARIABILITY OF PROPERTY VALUES, DAMAGE FACTORS, AND SO ON. IN ADDITION, THE INCREMENTAL SEISMIC RISK DUE TO THE PRESENCE OF NUCLEAR POWER PLANTS IS EVALUATED UTILIZING RESULTS AND METHODS AVAILABLE IN THE LITERATURE. THE RESULTS SHOW THAT THE INCREMENTAL RISK IS GENERALLY VERY SMALL COMPARED TO THE BACKGROUND SEISMIC RISK, EVEN IF A VERY HIGH PROBABILITY FOR CORE MELT IS POSTULATED. (EWH)
WORDS PROBABILITY;SEISMIC DESIGN;RISK;SEISMOLOGY;ANALYTICAL MODEL

/0/0000001-00000//// 50
SSION NO. 00C0155254
E AUTOMATED PRELIMINARY RISK ANALYSIS (AUTOET II)
HOR(S) WILSON JR
AUTH EG&G IDAHO INC., IDAHO FALLS
1978
L
CONF-780507-P3 +. 6 PPS, PP. XIII.5-1 THRU -6, FROM VOL. 3, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
L AVAILABILITY - JAMES R. WILSON, EG&G IDAHO INC., P.O. BOX 1625, IDAHO FALLS, IDAHO 83401
GORY 010000
TION 0104
CODE INE
TRY A
RACT AUTOET II IS A COMPUTER CODE WHICH AUTOMATICALLY DRAWS A SYSTEM EVENT TREE, ELIMINATING UNNECESSARY BRANCHES, AND LABELS EACH ACCIDENT SEQUENCE WITH PROBABILITY, CONSEQUENCES AND RISK. THE INPUTS TO THE CODE ARE KEY SUBSYSTEMS, CUT SETS (ON A SUBSYSTEM BASIS), CONSEQUENCES OF SUBSYSTEM FAILURE IN EACH FAILURE MODE, AND INITIATING EVENT PROBABILITY. (EWH)
ORDS PROBABILITY;FAULT TREE ANALYSIS;COMPUTER PROGRAM;RISK

0/0000001-00000//// 51
SSION NO. 00C0155244
E SYSTEM EVENT TREE ANALYSES FOR DETERMINING ACCIDENT SEQUENCES THAT DOMINATE RISKS IN LWR POWER PLANTS
OR(S) ASSELIN SV;CARLSON DD;FEDELE MA
AUTH SANDIA LABS., N.M. ; EVALUATION ASSOCIATES INC., PA.
1978
L
CONF-780507-P3 +. 12 PPS, PP. XII.2-1 THRU -12, FROM VOL. 3, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
L AVAILABILITY - MARIO A. FEDELE, EVALUATION ASSOCIATES INC., GSB BLDG., BALA CYNWYD, PA. 19004
GORY 050000;090000;170000
ION 0104
CODE AUA
TRY A
RACT A METHODOLOGY HAS BEEN DEVELOPED THAT IDENTIFIES THE DOMINANT ACCIDENT SEQUENCES FOR A GIVEN NUCLEAR POWER PLANT WITH SUBSTANTIALLY LESS EFFORT THAN WAS REQUIRED TO ACHIEVE THE SAME GOAL IN THE REACTOR SAFETY STUDY. THIS IS ACCOMPLISHED BY USING THE RSS RESULTS AND METHODS AS A POINT OF DEPARTURE, CAPITALIZING ON SIMILARITIES AMONG POWER PLANTS AND ACCOMMODATING THE DIFFERENCES IN PLANT SYSTEM FUNCTIONS AND INTERACTIONS. (EWH)
ORDS PROBABILITY;OPERATING EXPERIENCE;FAULT TREE ANALYSIS;ACCIDENT ANALYSIS;RISK;ANALYTICAL TECHNIQUE

SSION NO. 00C0153243
LE WASH-1400 INSIGHTS UTILIZED IN ASSESSING ALTERNATE CONTAINMENT DESIGNS
HOR(S) CARLSON DD;HICKMAN JW;TAYLOR MA
PAUTH SANDIA LABS., ALBUQUERQUE, N.M. ; U.S. NUCLEAR REGULATORY COMMISSION
E 1978
E L
D CONF-780507-P3 +. 16 PPS, PP. XII.1-1 THRU -16, FROM VOL. 3, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
IL AVAILABILITY - DAVID D. CARLSON, SANDIA LABS., ALBUQUERQUE, N.M. 87185
EGORY 110000
TION 0104
P CODE AOA;NRC
NTRY A
TRACT A VALUE-IMPACT ASSESSMENT IS MADE OF ALTERNATE CONTAINMENT CONCEPTS FOR COMMERCIAL LIGHT WATER REACTOR POWER PLANTS. SEVERAL ALTERNATE CONTAINMENT CONCEPTS ARE EVALUATED AND COMPARED CONSIDERING THEIR POTENTIAL FOR REDUCING PUBLIC RISK AND THEIR CONSTRUCTION COST. THE RESULTS AND METHODOLOGY OF THE REACTOR SAFETY STUDY (WASH-1400) ARE USED AS A BASIS FOR DETERMINING POTENTIAL RISK REDUCTIONS THAT COULD BE REALIZED BY ALTERNATE CONTAINMENT DESIGNS. AMONG THE ALTERNATIVES CONSIDERED, FILTERED ATMOSPHERIC VENTING APPEARS TO OFFER THE GREATEST POTENTIAL FOR REDUCING PUBLIC RISK FOR THE LEAST IMPACT. (EWH)
WORDS PROBABILITY;OPERATING EXPERIENCE;CONTAINMENT DESIGN;SAFETY EVALUATION;RISK;CONTAINMENT, PRESSURE VENTING

0/0000001-00000777/ 53
ESSION NO. 00C0153232
LE PROBABILISTIC RISK ANALYSIS OF RADIOACTIVITY RELEASE AND TRANSPORT FROM GEOLOGIC DISPOSAL OF RADIOACTIVE WASTES
HOR(S) BERTOZZI G;CARETTA A;SCHNEIDER H
PAUTH JRC ISPRA ESTABLISHMENT, ITALY
E 1978
E L
D CONF-780507-P3 +. 12 PPS, PP. IX.4-1 THRU -12, FROM VOL. 3, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
L AVAILABILITY - A. CARETTA, COMMISSION OF THE EUROPEAN COMMUNITIES, JOINT RESEARCH CENTRE, ISPRA ESTABLISHMENT, ITALY
EGORY 140000;130000
TION 0104
NTRY A
TRACT AN INTEGRATED MODELING SYSTEM FOR CALCULATING RADIONUCLIDE RELEASE AND BIOSPHERE TRANSPORT PROCESSES FROM GEOLOGIC DEPOSITORES TO MAN, IS PRESENTED. BECAUSE OF LARGE UNCERTAINTIES IN THE MAJOR INPUT DATA, A PROBABILISTIC APPROACH HAS BEEN FOLLOWED, GIVING THE RESULTS IN FORM OF HISTOGRAMS. THIS TECHNIQUE ALLOWS THE IDENTIFICATION OF THOSE PARAMETERS WHICH CONTROL THE MODEL RESULTS AND ON WHICH MAJOR EFFORT MAY BE FOCUSED. (EWH)
ORDS PROBABILITY;FUEL CYCLE;RISK;RADIOACTIVITY RELEASE;FISSION PRODUCT TRANSPORT;WASTE DISPOSAL, BEDROCK;WASTE DISPOSAL, TERRESTRIAL

0/0000001-00000777/ 54
SSION NO. 00C0153231
E COST-BENEFIT ESTIMATE OF TRANSPORTING SPENT NUCLEAR FUEL BY SPECIAL TRAINS
OR(S) GARRICK BJ;KAPLAN S
AUTH PICKARD, LOWE & GARRICK INC., IRVINE, CALIF.
1978
L
CONF-780507-P3 +. 10 PPS, PP. IX.3-1 THRU -10 FROM VOL. 3, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
L AVAILABILITY - STANLEY KAPLAN, PICKARD, LOWE & GARRICK INC.,

2070 BUSINESS CENTER DRIVE, SUITE 125, IRVINE, CALIF. 92715
EGORY 010000;030000
TION 0104
NTRY A
TRACT SPECIAL TRAINS HAVE BEEN ADVOCATED BY THE RAILROADS AS A MEANS OF REDUCING THE RISK ASSOCIATED WITH THE SHIPMENT OF RADIOACTIVE MATERIALS. SUCH TRAINS WOULD HAVE SEVERE LIMITATIONS ON SPEED, PASSING, CONDITIONS, AND TRAIN SIZE. THE RESULTING EXTRA COST FOR SPECIAL TRAINS IS ABOUT $20,000/AVERAGE SHIPMENT. THE QUESTION TO BE ANSWERED IS, "IS THIS MUCH REDUCTION IN RISK EVEN IF IT COULD BE ACHIEVED BY SPECIAL TRAINS, WORTH THE $20,000 PER SHIPMENT"? A PROBABILISTIC RISK ANALYSIS IS PERFORMED. (EWH)
WORDS PROBABILITY;FUEL CYCLE;COST BENEFIT;TRANSPORTATION AND HANDLING; FUEL, NUCLEAR;INDUSTRY, TRANSPORTATION;RISK

/07/0000001-00000777 55
ESSION NO. 000153230
LE SOME ASPECTS OF THE RISKS ASSOCIATED WITH A MIXED OXIDE FUEL PRODUCTION PLANT
HOR(S) CANDOLFO G;LOMAZZI F;CARETTA A;ROCCO P
PAUTH AGIP NUCLEARE, MILANO, ITALY ; JRC ISPRA ESTABLISHMENT, ITALY
E 1978
E L
O CONF-780507-P3 +. 12 PPS. PP. IX.2-1 THRU -12, FROM VOL. 3, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
IL AVAILABILITY - GIOIA CANDOLFO, AGIP NUCLEARE, C.P. 1629, I-20122 MILANO, ITALY
EGORY 130000
TION 0104
NTRY A
TRACT THE RISKS ASSOCIATED WITH A MIXED OXIDE FUEL FABRICATION PLANT ARE REVIEWED. THE STUDY FOCUSES ON A DETAILED PROBABILISTIC ANALYSIS OF ONE OF THE MAJOR CONCEIVABLE INTERNAL ACCIDENTS: FIRE IN A HOT CELL. THE PROBABILISTIC METHODOLOGY USED CALLS FOR A DEVELOPMENT OF THE ACCIDENT EVENT TREE WITH THE RELATED ANALYSIS OF THE INTERVENING SYSTEM FAULT-TREES; MOREOVER, A SPECTRUM OF POSSIBLE FIRE MAGNITUDES IS ASSUMED AND FOR EACH FIRE EVENT THE PU-RELEASE AT THE STACK IS EVALUATED BY AN ORIGINAL COMPUTER MODEL. (EWH)
WORDS PROBABILITY;FUEL CYCLE;RISK;MIXED OXIDE;PRODUCTION;FUEL, NUCLEAR;FABRICATION;FAULT TREE ANALYSIS;ANALYTICAL MODEL

/07/0000001-00000777 56
ESSION NO. 000153229
LE DEVELOPMENT OF RISK ASSESSMENT METHODOLOGY APPLICABLE TO RADIOACTIVE WASTE ISOLATION
HOR(S) CAMPBELL JE;MCGRATH PE;CULLINGFORD MC
PAUTH SANDIA LABS., ALBUQUERQUE, N.M. ; U.S. NUCLEAR REGULATORY COMMISSION
1978
L
CONF-780507-P3 +. 16 PPS. PP. IX.1-1 THRU -16, FROM VOL. 3, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
L AVAILABILITY - P.E. MCGRATH, SANDIA LABS., ALBUQUERQUE, N.M.
GORY 140000;130000
ION 0104
CODE AOA;NRC
TRY A
RACT THE RISK FROM RADIOACTIVE WASTE DISPOSAL IN A DEEP GEOLOGIC FORMATION HAS NOT YET BEEN COMPLETELY ASSESSED. A COMPLETE ASSESSMENT SHOULD INCLUDE CREDIBLE ESTIMATES OF THE LIKELIHOOD THAT RADIOACTIVE MATERIALS WOULD ESCAPE THE REPOSITORY AND ENTER THE HUMAN ENVIRONMENT, AND THE MAGNITUDE OF THE RESULTANT CONSEQUENCES IN TERMS OF HUMAN HEALTH EFFECTS. IN ADDITION, SUCH AN ASSESSMENT SHOULD IDENTIFY THE DOMINANT CONTRIBUTORS TO RISK AND, TO THE EXTENT POSSIBLE, QUANTIFY THE UNCERTAINTIES IN RISK ESTIMATES. (EWH)
URDS PROBABILITY;FUEL CYCLE;RISK;ANALYTICAL TECHNIQUE;WASTE MANAGEMENT;WASTE DISPOSAL;CONTAINMENT ISOLATION

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SSION NO.   00C0153228
LE          ESTIMATION OF NUCLEAR POWER PLANT AIRCRAFT HAZARDS
HOR(S)      GOTTLIEB P
PAUTH       DAMES & MOORE, LOS ANGELES, CALIF.
E           1978
E           L
U           CONF-780507-P3 +.  8 PPS, PP. VIII.8-1 THRU -8, FROM VOL. 3,
            PROCEEDINGS OF    THE ANS TOPICAL MEETING ON PROBABILISTIC
            ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY
            8-10, 1978
IL          AVAILABILITY - PETER GOTTLIEB, DAMES & MOORE, 1100 GLENDON
            AVE., LOS ANGELES, CALIF.  90024
EGORY       020000
TION        0104
P CODE      382
NTRY        A
TRACT       THE STANDARD PROCEDURES FOR ESTIMATING AIRCRAFT RISK TO NUCLEAR
            POWER PLANTS PROVIDE A CONSERVATIVE ESTIMATE, WHICH IS ADEQUATE
            FOR MOST SITES, WHICH ARE NOT CLOSE TO AIRPORTS OR HEAVILY
            TRAVELED AIR CORRIDORS.  IN MANY INSTANCES THE VERY LARGE
            COMMERCIAL AIRCRAFT CAN BE SHOWN TO HAVE AN ACCEPTABLY SMALL
            IMPACT FREQUENCY, WHILE THE VERY SMALL GENERAL AVIATION
            AIRCRAFT WILL NOT PRODUCE SUFFICIENTLY SERIOUS IMPACT TO IMPAIR
            THE SAFETY-RELATED FUNCTIONS.  THIS PAPER EXAMINES THE IN
            BETWEEN AIRCRAFT:  PRIMARILY TWIN-ENGINE, USED FOR BUSINESS,
            PLEASURE, AND AIR TAXI OPERATIONS.  (EWH)
WORDS       PROBABILITY;ANALYTICAL TECHNIQUE;AIRCRAFT;RISK;ACCIDENT ANALYSIS
/0/0000001-000007777          58
SSION NO.   00C0153227
LE          PROBABILISTIC EVALUATION OF RISKS ASSOCIATED WITH AVIATION,
            ROAD, RAILWAY, AND RIVER TRAFFIC ADJACENT TO NUCLEAR POWER
            PLANTS IN FRANCE
HOR(S)      PROCACCIA H;GOBERT T
PAUTH       ELECTRICITE DE FRANCE
E           1978
E           L
U           CONF-780507-P3 +.  12 PPS, PP. VIII.7-1 THRU -12, FROM VOL. 3,
            PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC
            ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY
            8-10, 1978
IL          AVAILABILITY - HENRI PROCACCIA, ELECTRICITE DE FRANCE, RESEARCH
            & DEVELOPMENT OFFICE, 25, ALLEE PRIVEE-CARREFOUR PLEYEL, 93206
            SAINT-DENIS CEDEX, FRANCE
EGORY       020000;030000
TION        0104
P CODE      EDF
TRY         A
TRACT       ELECTRICITE DE FRANCE HAS DEVELOPED A METHODOLOGY TO CALCULATE
            THE POTENTIAL IMPACT OF RISKS TO A NUCLEAR POWER PLANT
            SUBSEQUENT TO AN ACCIDENT OCCURRING ON TRANSPORTATION NETWORKS
            IN THE VICINITY OF A NUCLEAR SITE.  RISKS ENTAILED IN AVIATION,
            AND IN ROAD, RAIL, AND RIVER TRAFFIC HAVE BEEN ESTIMATED.  THIS
            METHODOLOGY HAS BEEN APPLIED TO ALL SITES PLANNED IN FRANCE.
            (EWH)
ORDS        PROBABILITY;ANALYTICAL TECHNIQUE;FRANCE;RISK;SITING;AIRCRAFT;
            ACCIDENT ANALYSIS
/0/0000001-000007777          59
SSION NO.   00C0153226
LE          TORNADO MISSILE SIMULATION AND RISK ANALYSIS
OR(S)       TWISDALE LA;CHU J;DUNN WL
AUTH        CAROLINA POWER & LIGHT CO., RALEIGH, N.C. ; N.C. STATE UNIV.,
            RALEIGH
            1978
            L
            CONF-780507-P3 +.  15 PPS, PP. VIII.6-1 THRU -15, FROM VOL. 3,
            PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC
            ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY
            8-10, 1978
L           AVAILABILITY - L.A. TWISDALE, CAROLINA POWER & LIGHT, P.O. BOX
            1551, RALEIGH, N.C. 27602
GORY        160000;050000
ION         0104
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P CODE APY;NCU
NTRY A
TRACT MATHEMATICAL MODELS OF THE CONTRIBUTING EVENTS TO THE TORNADO MISSILE HAZARD AT NUCLEAR POWER PLANTS HAVE BEEN DEVELOPED IN WHICH THE MAJOR SOURCES OF UNCERTAINTY HAVE BEEN CONSIDERED IN A PROBABILISTIC FRAMEWORK. THESE MODELS HAVE BEEN STRUCTURED INTO A SEQUENTIAL EVENT FORMALISM WHICH PERMITS THE TREATMENT OF BOTH SINGLE AND MULTIPLE MISSILE GENERATION EVENTS. A SIMULATION COMPUTER CODE UTILIZING THESE MODELS HAS BEEN DEVELOPED TO OBTAIN ESTIMATES OF TORNADO MISSILE EVENT LIKELIHOODS. TWO CASE STUDIES HAVE BEEN ANALYZED. (EWH)
YWORDS PROBABILITY;ANALYTICAL TECHNIQUE;MISSILE GENERATION AND PROTECTION;DESTRUCTIVE WIND;SIMULATION;RISK;ANALYTICAL MODEL

/0/0000001-000007772 60
ESSION NO. 0000155083
LE THE USE OF QUANTITATIVE RISK AND PROBABILISTIC SAFETY CRITERIA IN THE CONCEPTUAL DESIGN OF A LARGE POOL-TYPE LMFBR
HOR(S) HARTUNG JA;LANCET RT
PAUTH ATOMICS' INTERNATIONAL, CANOGA PARK, CALIF.
L 1978
E L
U CONF-780507-P2 +. 10 PPS. PP. VII.4-1 THRU -10. FROM VOL. 2. PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
IL AVAILABILITY - ROBERT T. LANCET, ATOMICS INTERNATIONAL, ROCKWELL INTERNATIONAL CORP., 8900 DE SOTO AVE., CANOGA PARK, CALIF. 91304
EGORY 010000;160000
TION 0104
P CODE CAE
NTRY A
TRACT METHODOLOGY IS SUMMARIZED WHICH DEFINES A SET OF PLANT CONDITIONS (BASED ON PROBABILITY OF OCCURRENCE) AND CORRESPONDING SAFETY CRITERIA. THE MAJOR DIFFERENCE BETWEEN THIS METHODOLOGY AND A MORE CONVENTIONAL (DBA) SAFETY APPROACH IS THE ADDITION OF THE PLANT CONDITION CATEGORY FOR RESIDUAL RISK EVENTS. THESE ARE DEFINED AS EVENTS WHICH ARE LESS LIKELY TO OCCUR THAN DBA'S BUT WHICH HAVE POTENTIAL CONSEQUENCES HIGH ENOUGH TO MERIT EXPLICIT CONSIDERATION ON A RESIDUAL RISK BASIS. ONLY CORE DISRUPTIVE ACCIDENT RELATED EVENTS HAVE BEEN INCLUDED IN THIS CATEGORY TO DATE. (EWH)
WORDS PROBABILITY;DESIGN CRITERIA;RISK;SAFETY ANALYSIS;DESIGN STUDY; REACTOR, LMFBR;REACTOR, POOL TYPE

/0/0000001-000007772 61
ESSION NO. 0000155082
E RISK ALLOCATION APPROACH TO REACTOR SAFETY DESIGN AND EVALUATION
OR(S) GOKCEK O;TEMME I;DERBY SL
AUTH GENERAL ELECTRIC CO., SUNNYVALE, CALIF.
1978
L
CONF-780507-P2 +. 12 PPS. PP. VII.3-1 THRU -12. FROM VOL. 2. PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
L AVAILABILITY - MARK I. TEMME, GENERAL ELECTRIC CO., FAST BREEDER REACTOR DEPT.,310 DE GUIGNE DRIVE, SUNNYVALE, CALIF. 94086
GORY 010000;180000
ION 0104
CODE GEC
TRY A
RACT DESCRIBES A RISK ALLOCATION TECHNIQUE USED FOR DETERMINING NUCLEAR POWER PLANT DESIGN RELIABILITY REQUIREMENTS. THE CONCEPT OF RISK ALLOCATION - OPTIMUM CHOICE OF SAFETY FUNCTION RELIABILITIES UNDER A MAXIMUM RISK CONSTRAINT - IS DESCRIBED. AN EXAMPLE OF RISK ALLOCATION IS PRESENTED TO DEMONSTRATE THE APPLICATION OF THE METHODOLOGY. (EWH)
RDS PROBABILITY;DESIGN CRITERIA;RISK;SAFETY EVALUATION;DESIGN STUDY; POWER PLANT, NUCLEAR

SSION NO. 00C0153061
LE THE ROLE OF RISK CRITERIA IN NUCLEAR PLANT DECISIONS
HOR(S) TEMME MI;MCDONALD A
PAUTH GENERAL ELECTRIC CO., SUNNYVALE, CALIF.
E 1978
E L
O CONF-780507-B +. 12 PPS, PP. VII.2-1 THRU -12, FROM VOL. 2, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
IL AVAILABILITY - M.I. TEMME, GENERAL ELECTRIC CO., FAST BREEDER REACTOR DEPT. 310 DE GUIGNE DR., P.O. BOX 5020, SUNNYVALE, CALIF. 94086
EGORY 010000
ITION 0104
P CODE GEC
NTRY A
STRACT REQUIREMENTS FOR THE DETERMINATION OF RISK ACCEPTANCE CRITERIA ARE DERIVED FROM REVIEWS OF PAST EFFORTS RELATED TO THEIR DEVELOPMENT AND FROM SOCIAL DECISION THEORY. IT IS OBSERVED THAT ATTEMPTS TO OBJECTIVELY DETERMINE A DEFINITIVE SET OF SOCIETAL RISK ACCEPTANCE CRITERIA CANNOT SUCCEED BECAUSE THERE IS NO OBJECTIVE PROCEDURE FOR AGGREGATING THE VALUES AND PREFERENCES OF INDIVIDUALS INTO AN EXPRESSION THAT CAN BE CALLED "SOCIETY'S PREFERENCES". THE AUTHORS SUGGEST THAT EFFORTS BE FOCUSED ON DETERMINING "MAXIMUM RISK CRITERIA" (WITHOUT THE CONNOTATION OF SOCIETAL ACCEPTANCE). SOME THOUGHTS ON THE PROPER ROLE OF SUCH CRITERIA ARE EXPRESSED, AND THE METHOD OF MULTIATTRIBUTE UTILITY THEORY FOR THEIR DETERMINATION IS DESCRIBED. (EWH)
WORDS PROBABILITY;DESIGN CRITERIA;RISK;SOCIO/PHILOSOPHICAL CONSIDERATION

/0/0000001-00000077// 63
SSION NO. 00C0153069
LE RISK ASSESSMENT METHODS APPLICATION TO THE FAST FLUX TEST FACILITY
HOR(S) FELLER K
PAUTH GENERAL ELECTRIC CO., SUNNYVALE, CALIF.
E 1978
E L
O CONF-780507-P2 +. 12 PPS, PP. V.4-1 THRU -12, FROM VOL. 2, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
IL AVAILABILITY - KIM FELLER, GENERAL ELECTRIC CO., 310 DEGUIGNE DRIVE, SUNNYVALE, CALIF. 94086
EGORY 180000;010000
TION 0104
P CODE GEC
NTRY A
TRACT THE TECHNICAL APPROACH AND RESULTS OF A RISK ASSESSMENT STUDY BASED ON THE FAST FLUX TEST FACILITY REFERENCE DESIGN ARE SUMMARIZED. THE STUDY WAS CONDUCTED AS PART OF AN ONGOING EFFORT TO DEVELOP A TECHNOLOGY BASE FOR BREEDER REACTOR RISK ASSESSMENT. THE PRIMARY STUDY OBJECTIVE WAS METHODS DEVELOPMENT. DUE TO THE LIMITED SCOPE AND LACK OF EXPERT PARTICIPATION FROM MANY IMPORTANT TECHNOLOGY AREAS, THE RESULTS ARE NOT INTENDED TO REPRESENT THE RISK RESULTING FROM FFTF OPERATION. (EWH)
ORDS PROBABILITY;FFTF (TR);SAFETY ANALYSIS;RISK;REACTOR, BREEDER; DESIGN STUDY

0/0000001-00000077// 64
SSION NO. 00C0153041
E CONTAINMENT DESIGN OPTIONS FOR THE HTGR: AN APPLICATION OF PROBABILISTIC RISK ASSESSMENT
OR(S) BARSELL AW;ORVIS DD
AUTH GENERAL ATOMIC CO., SAN DIEGO, CALIF.
1978
L
CONF-780507-P1 +. 12 PPS, PP. V.7-1 THRU -12, FROM VOL. 1, PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY

8-10, 1978
IL AVAILABILITY - ARTHUR W. BARSELL, HTGR SAFETY BRANCH, GENERAL ATOMIC CO., P.O. BOX 81608, SAN DIEGO, CALIF. 92138
EGORY 110000;010000
TION 0104
P CODE LAW
NTRY A
TRACT AN EVALUTION OF TWO ALTERNATIVES TO CONVENTIONAL STEEL-LINED CLOSED CONTAINMENT HAS BEEN PERFORMED FOR A 3000 MW(T) HIGH TEMPERATURE GAS COOLED REACTOR THROUGH APPLICATION OF PROBABILISTIC RISK ASSESSMENT. THE EVALUATION IS BASED ON ADAPTATION OF AN EXTENSIVE PRA PERFORMED FOR CONVENTIONAL CONTAINMENT AND IS BELIEVED TO BE INDICATIVE OF WHAT MAY BE ACHIEVED. (EWH)
YWORDS PROBABILITY;CONTAINMENT DESIGN;REACTOR, HTGR;RISK;SAFETY EVALUATION;DESIGN STUDY;ECONOMIC STUDY

070000001-00000777 65
ESSION NO. 00C0153040
LE PUBLIC ATTITUDES IN RELATION TO THE RISKS PRESENTED BY NEW TECHNOLOGIES
HOR(S) CAVE L;HOLMES RE;HOLMES PJ
PAUTH UNIV. OF CALIF., LOS ANGELES ; POLLUTION PREVENTION LTD., SUSSEX, U.K.
E 1978
E L
D CONF-780507-P1 +. 12 PPS. PP. IV.8-1 THRU -12, FROM VOL. 1. PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
IL AVAILABILITY - L. CAVE, SCHOOL OF ENGINEERING AND APPLIED SCIENCE, UNIV. OF CALIF., LOS ANGELES, CALIF.
EGORY 010000
TION 0104
P CODE UAV
NTRY A
TRACT STUDIES ON SOCIETAL ATTITUDES DURING THE NINETEENTH CENTURY TO RISKS INTRODUCED BY NEW TECHNOLOGIES INDICATE THAT THE PRESENT DAY RELUCTANCE ON THE PART OF SOCIETY TO ACCEPT A LEVEL OF RISK, WHICH ON A NATIONAL BASIS APPEARS TO BE INSIGNIFICANT, IN ORDER TO HAVE THE BENEFITS OF A NEW TECHNOLOGY AT A REASONABLE COST, MAY NOT BE A NEW PHENOMENON. CONDITIONS ARE POSTULATED IN WHICH HISTORICAL DATA ON PUBLIC ATTITUDES TO RISK ARE VALID, AND THE APPLICATION OF THIS APPROACH IS ILLUSTRATED BY A STUDY OF SOCIETAL ATTITUDES IN THE NINETEENTH CENTURY TO THE RISKS PRESENTED BY THE NEW TECHNOLOGY OF RAILWAY TRAVEL IN THE UNITED KINGDOM. (EWH)
WORDS PROBABILITY;TECHNOLOGY;RISK;BENEFIT VS RISK;UNITED KINGDOM; SOCIO/PHILOSOPHICAL CONSIDERATION

070000001-00000777 66
ESSION NO. 00C0153037
LE THE NEED FOR GREATER INTER-AGENCY COOPERATION, AND CONTROLLING RISKS ASSOCIATED WITH NON-NUCLEAR SYSTEMS
HOR(S) CLIFFORD PR
PAUTH THE MITRE CORP., MCLEAN, VA.
1978
L
CONF-780507-P1 +. 11 PPS. PP. XI.4-1 THRU -11, FROM VOL. 1. PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
L AVAILABILITY - PAUL R. CLIFFORD, THE MITRE CORP., 1820 DOLLEY MADISON BLVD., MCLEAN, VA. 22101
GORY 010000
ION 0104
CODE ALW
TRY A
RACT PRESENTS AN OVERVIEW OF RISKS ASSOCIATED WITH ALL NON-NUCLEAR ENERGY SYSTEMS AND THE NEED FOR INTER-AGENCY COOPERATION. STATES THAT THERE IS A NEED FOR INCREASED COOPERATION AMONG THE FEDERAL AGENCIES IN THE NON-NUCLEAR ENERGY FIELD TO ENSURE THE SAFETY OF WORKERS AND THE PUBLIC. PRECEDENTS FOR INTER-AGENCY COOPERATION ARE BEING ESTABLISHED IN THE FIELD OF CONTROL OF

TOXIC SUBSTANCES. THESE PRECEDENTS SHOULD BE FOLLOWED IN REGULATING NON-NUCLEAR ENERGY SYSTEMS. (EWH)
WORDS PROBABILITY;RISK;BENEFIT VS RISK;ENERGY SOURCE;TECHNOLOGY; TOXICITY;COMPARISON;ACCIDENT

/070000001-00000777/ 67
ESSION NO. 00C0153036
LE PUBLIC RECEPTION FOR NUCLEAR ENERGY
HOR(S) LAVE L
PAUTH CARNEGIE-MELLON UNIV., PITTSBURGH, PA.
E 1978
E L
D CONF-780507-P1 +. 8 PPS. PP. XI.3-1 THRU -8. FROM VOL. 1. PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
IL AVAILABILITY - LESTER LAVE, GSIA, CARNEGIE-MELLON UNIV., PITTSBURGH, PA. 15213
EGORY 010000
TION 0104
NTRY A
TRACT THE FOLLOWING TOPICS ARE DISCUSSED: WHY ARE NUCLEAR REACTORS UNDER ATTACK?, NORMATIVE THEORIES OF DECISION-MAKING, AND BEHAVIOR CONCERNING RISKY SITUATIONS.
WORDS PROBABILITY;BENEFIT VS RISK;RISK;SOCIO/PHILOSOPHICAL CONSIDERATION;N-POWER FORECAST;INDUSTRY, PROBLEM/PROPOSAL

/070000001-00000777/ 68
ESSION NO. 00C0153035
LE THE SOCIO-ECONOMIC ASPECTS OF RISK-BENEFIT DECISION MAKING
HOR(S) WHIPPLE C
PAUTH ELECTRIC POWER RESEARCH INST., PALO ALTO, CALIF.
E 1978
E L
D CONF-780507-P1 +. 4 PPS. PP. XI.2-1 THRU -4. FROM VOL. 1. PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
IL AVAILABILITY - CHRIS WHIPPLE, ELECTRIC POWER RESEARCH INST., 3412 HILLVIEW AVE., PALO ALTO, CALIF. 94304
EGORY 010000
TION 0104
CODE EPR
NTRY A
TRACT THREE APPLICATIONS FOR ESTIMATES OF RISKS ARE: 1) DECIDING WHERE TO SPEND MONEY TO BUY A LITTLE MORE SAFETY AND WHERE IT IS COST EFFECTIVE; 2) CHOOSING BETWEEN TECHNICAL ALTERNATIVES; OR, 3) DECIDING WHICH TECHNOLOGIES TO HAVE AND WHICH TO NOT HAVE. (EWH)
WORDS MEASUREMENT;PROBABILITY;RISK;BENEFIT VS RISK; SOCIO/PHILOSOPHICAL CONSIDERATION

/070000001-00000777/ 69
ESSION NO. 00C0153034
LE SAFETY INVESTIGATION OF A MAJOR PETROCHEMICAL COMPLEX
HOR(S) FARMER FR
PAUTH UNITED KINGDOM ATOMIC ENERGY AUTHORITY, WARRINGTON
1978
L
CONF-780507-P1 +. 13 PPS. PP. XI.1-1 THRU -13. FROM VOL. 1. PROCEEDINGS OF THE ANS TOPICAL MEETING ON PROBABILISTIC ANALYSIS OF NUCLEAR REACTOR SAFETY; NEWPORT BEACH, CALIF., MAY 8-10, 1978
L AVAILABILITY - F.R. FARMER, UNITED KINGDOM ATOMIC ENERGY AUTHORITY, WIGSHAW LANE, CULCHETH WARRINGTON WA3 4NE, ENGLAND
GORY 010000
ION 0104
CODE UKA
TRY A
RACT DISCUSSES THE UNITED KINGDOM'S EXPERIENCES, ORGANIZATIONS, AND STUDIES CONCERNING THE RISK TO THE POPULATION FOR VARIOUS KEYWORDS ASSOCIATED WITH AN INDUSTRIALIZED NATION. THESE ARE PLACED IN COMPARATIVE PERSPECTIVE AND WEIGHED AGAINST PREVAILING CRITERIA. (EWH)

WORDS PROBABILITY;SAFETY EVALUATION;REVIEW;RISK;BENEFIT VS RISK; SITING, CHEMICAL PROCESS PLANT

07000001-00007777 70
ESSION NO. 00RC151593
LE RISK ANALYSIS METHODS DEVELOPMENT, EIGHTH QUARTERLY REPORT OCTOBER-DECEMBER 1978
RPAUTH GENERAL ELECTRIC CO., SUNNYVALE, CALIF.
TE 1979
PE C
MO GEFR-14023-6 +. 22 PPS, 3 REFS, JAN. 1979
AIL AVAILABILITY - LIMITATIONS ON DISTRIBUTION; SEND REQUESTS TO DOE TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830
EGORY 010000;090000
TION 0100
RP CODE GEC
NTRY A
TRACT PROGRAM PLANNING FOR FISCAL YEAR 1979 WAS COMPLETED. MAJOR ACTIVITIES WILL INCLUDE: (1) COMPLETION OF THE LOA-2 TOP PILOT STUDY; (2) COMPLETION OF AN LOA-2 STUDY OF THE LOPI SEQUENCE, INCLUDING DEVELOPMENT OF THE STATISTICAL SUCCESS MODEL AND STATISTICAL ANALYSIS OF EXISTING AND PROPOSED TEST RESULTS; (3) ISSUANCE OF THE PROCEDURAL MANUAL FOR BREEDER REACTOR RISK ASSESSMENT; AND (4) CONTINUED LEVEL-OF-EFFORT SUPPORT OF LOA WORKING GROUPS.(FAH)
WORDS RISK;ANALYTICAL TECHNIQUE;REACTOR, LMFBR;R AND D PROGRAM; PROBABILITY;ANALYTICAL MODEL

07000001-00007777 71
ESSION NO. 00J0150615
LE ON THE USE OF A BAYESIAN REASONING IN SAFETY AND RELIABILITY DECISIONS - THREE EXAMPLES
HOR(S) KAPLAN S;GARRICK BJ
PAUTH PICKARD, LOWE & GARRICK INC., IRVINE, CALIF.
E 1979
E D
O 14 PPS, 6 TABS, 12 FIGS, 2 REFS, NUCLEAR TECHNOLOGY, 44(2), PP. 231-45 (JULY 1979)
EGORY 030000;010000
TION 0098
NTRY A
BB NUAT
TRACT BAYES' THEOREM IS USED TO QUANTIFY THE IMPACT OF "NEW EVIDENCE" IN THREE ENERGY-RELATED DECISION PROBLEMS. THE FIRST PROBLEM CONCERNS THE RISK OF RADIOACTIVITY RELEASE DURING THE RAILROAD TRANSPORT OF SPENT NUCLEAR FUEL. THIS HISTORY OF SHIPMENTS THUS FAR IS SHOWN TO MAKE IT HIGHLY UNLIKELY THAT THE FREQUENCY OF RELEASE IS ON THE ORDER OF 10(EXP-3) OR GREATER PER SHIPMENT. THE SECOND AND THIRD APPLICATIONS INVOLVE PREDICTING THE AVAILABILITY PERFORMANCE OF NEW GENERATIONS OF TURBINE BLADES. BAYES' THEOREM IS DEMONSTRATED AS A MEANS FOR INCORPORATING IN THE PREDICTION THE LIMITED OPERATIONAL DATA ON THE NEW BLADES ALONG WITH THE EXPERIENCE OF THE EARLIER GENERATION AND THE KNOWLEDGE OF THE DESIGN CHANGES.
WORDS TRANSPORTATION AND HANDLING;TRAIN;CONTAINMENT ANALYSIS;SAFETY REVIEW;PROBABILITY;RISK;FUEL, NUCLEAR

07000001-00007777 72
ESSION NO. 00X0150591
LE PROBABILITY INTERVALS FOR THE RELIABILITY OF COMPLEX SYSTEMS USING MONTE CARLO SIMULATION
HOR(S) LEE TY;SALEM SL
PAUTH UNIV. OF CALIF., LOS ANGELES
1977
N
PB-280191 + UCLA-ENG-7758 +. 45 PPS, 2 TABS, 3 FIGS, DEC. 1977
L AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GORY 010000
ION 0098
CODE UAV
TRY A
RACT A METHOD OF ASSESSING THE UNCERTAINTY IN THE TOP EVENT

PROBABILITY OF A FAULT TREE BY A MONTE CARLO SIMULATION IS PRESENTED IN THIS REPORT. THIS APPROACH IS USED TO PRODUCE AN EMPIRICAL BAYES ESTIMATE OF THE PROBABILITY INTERVALS OF TOP EVENT UNRELIABILITY. THE MONTE CARLO SIMULATION USED HERE PROCEEDS BY PROPAGATING BASIC EVENT PROBABILITIES, CHOSEN RANDOMLY FROM INPUT PROBABILITY DISTRIBUTIONS THROUGH THE TREE, PRODUCING AN EMPIRICAL TOP EVENT PROBABILITY DISTRIBUTION (WITH ASSOCIATED CONFIDENCE LIMITS) AFTER COMBINING THE RESULTS OF MANY TRIALS. THIS REPORT WILL DISCUSS THE BASIC PRINCIPLES OF THIS MONTE CARLO APPROACH, AND WILL PRESENT A NEW COMPUTER CODE, LIMITS, WHICH IS THEN USED IN SEVERAL EXAMPLES.

YWORDS RELIABILITY, SYSTEM;MONTE CARLO;ANALYTICAL MODEL;SIMULATION; PROBABILITY;FAULT TREE ANALYSIS;RISK

3/0/0000001-0000077// 73

CESSION NO. 00X0150532
LE PROBABILISTIC SAFETY ANALYSIS IV
THOR(S) ERDMANN RC
RPAUTH SCIENCE APPLICATIONS INC., PALO ALTO, CALIF. (PREPARED FOR EPRI)
TL 1979
PE N
HO EPRI-NP-1039 +. 75 PPS, 19 TABS, 14 FIGS, APRIL 1979
IL AVAILABILITY - RECORDS & REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10412, PALO ALTO, CALIF. 94303
EGORY 090000;010000
TION 0098
P CODE SAI
NTRY A
TRACT SUMMARIZES WORK ACCOMPLISHED DURING 1977-78 ON EPRI CONTRACT RP767-1. THE PRIMARY GOAL OF THIS STUDY HAS BEEN TO CONTINUE DEVELOPMENT OF PROBABILISTIC RISK ASSESSMENT METHODOLOGY AND ITS APPLICABILITY TO UTILITY REQUIREMENTS. IMPROVEMENTS HAVE BEEN DEVELOPED FOR METHODS OF FAULT TREE ANALYSIS, CONSEQUENCE ANALYSIS, AND ATWS DATA COLLECTION AND RETRIEVAL. PROGRESS HAS BEEN DOCUMENTED FOR AN ANALYSIS OF EXTERNAL FUEL CYCLE RISKS. A TECHNIQUE HAS BEEN DEVELOPED FOR APPLYING REGIONAL SEISMIC HISTORIES TO ANALYSES OF RISK FROM EARTHQUAKES. (EWH)
WORDS EPRI;PROBABILITY;FAULT TREE ANALYSIS;SEISMIC DESIGN;RELIABILITY ANALYSIS;ACCIDENT, ATWS;FUEL CYCLE;SEISMOLOGY;ANALYTICAL TECHNIQUE;RISK;SAFETY ANALYSIS

/0/0000001-0000077// 74

SSION NO. 00C0148980
E ACTIVITIES OF THE COMMISSION OF THE EUROPEAN COMMUNITIES AND A CRITICAL REVIEW OF QUANTITATIVE CONCEPTS IN RISK ASSESSMENT
HOR(S) VAN REIJEN G;VINCK W
PAUTH COMMISSION OF THE EUROPEAN COMMUNITIES
1978
L
16 PPS, PAPER PRESENTED AT ANS INTERNATIONAL TOPICAL MEETING ON PROBABILISTIC ANALYSIS ON NUCLEAR REACTOR SAFETY; LOS ANGELES, CALIF., MAY 8-10, 1978
IL AVAILABILITY - W. VINCK, COMMISSION OF THE EUROPEAN COMMUNITIES, NUCLEAR SAFETY DIVISION, WETSTRAAT 200, 1049 BRUSSELS
EGORY 010000
TION 0095
P CODE EAB
NTRY L
RACT THIS PAPER GIVES A GENERAL SURVEY OF THE ACTIVITIES OF THE COMMISSION OF THE EUROPEAN COMMUNITIES IN HAZARD PREVENTION. THE SECOND PART DESCRIBES THE SITUATION WITH REGARD TO THE APPLICATION OF QUANTITATIVE RISK CONCEPTS IN NUCLEAR AND NON-NUCLEAR LICENSING PROCESSES. THE PAPER CONCLUDES WITH A CRITICAL REVIEW OF QUANTITATIVE ASSESSMENT AND ITS USE AND WITH CONSIDERATIONS IN FURTHER DEVELOPMENT OF THE METHODOLOGY. (FAH)
ORDS RISK;HAZARDS ANALYSIS;PROBABILITY;LICENSING PROCESS

0/0000001-0000077// 75

SSION NO. 00C0147869
E RISK-ASSESSMENT TECHNIQUES AND THE REACTOR LICENSING PROCESS
OR(S) LEVINE S
AUTH U.S. NUCLEAR REGULATORY COMMISSION
1979

L
CONF-780819 (VOL. 1)-- +. 24 PPS, PP. 8-31. FROM PROCEEDINGS OF THE 15TH DOE NUCLEAR AIR CLEANING CONFERENCE; BOSTON, MASS., AUG. 7-10, 1978
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 010000;070000;180?00
TION 0093
RP CODE NRC
UNTRY A
STRACT DESCRIBES THE NRC'S EFFORTS TO STABILIZE THE REACTOR LICENSING PROCESS. DESCRIBES THE REACTOR SAFETY STUDY (WASH-1400), CONCENTRATING ON THE ENGINEERING ASPECTS OF THE CONTRIBUTION TO REACTOR ACCIDENT RISKS. FINALLY DESCRIBES SOME NEW WORK BEGUN ON THE APPLICATION OF RISK-ASSESSMENT TECHNIQUES TO STABILIZE THE REACTOR LICENSING PROCESS.
WORDS ACCIDENT;REGULATION;LICENSING PROCESS;RISK;PROBABILITY;SAFETY PRINCIPLES AND PHILOSOPHY;ACCIDENT, PROBABILITY OF;DATA PROCESSING;AIR CLEANING

/0/0000001-00000777/ 76
ESSION NO. 0000147175
LE SELECTION OF EVENTS FOR A PROBABILISTIC EVALUATION OF PWR SAFETY (IN ENGLISH)
HOR(S) NAMY P
PAUTH FRAMATOME, FRANCE
E 1979
E L
D FRRSR-181 +. 8 PPS, FROM HAMBOURG CONFERENCE; MAY 6-9, 1979
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 010000;180000
TION 0092
NTRY F
TRACT THIS PAPER PRESENTS A METHOD WHICH CAN BE USEFULLY FOLLOWED TO SELECT INITIATING EVENTS TO BE RETAINED FOR A RISK ANALYSIS OF A NUCLEAR POWER PLANT. THE MAIN STEPS ARE THE FOLLOWING: 1. DETERMINATION AND JUSTIFICATION OF THE INITIATING EVENTS CHOSEN. 2. QUANTIFICATION AND RELIABILITY ANALYSIS OF ACCIDENT SEQUENCES INDUCED BY THE INITIATING EVENTS. 3. RADIOLOGICAL ANALYSIS OF THESE ACCIDENT SEQUENCES. THIS PAPER PRESENTS A GENERAL METHOD OF SELECTION WHICH HAS BEEN USED IN THE LICENSING PROCESS OF KOEBERG NUCLEAR POWER PLANT TO ANSWER THE FIRST STEP OF THE RISK ANALYSIS.
WORDS FRANCE;RISK;ANALYTICAL TECHNIQUE;LICENSING PROCESS;REACTOR, PWR; ACCIDENT, PROBABILITY OF;PROBABILITY;FOREIGN EXCHANGE

/0/0000001-00000777/ 77
ESSION NO. 0000144624
LE REPORT OF THE NRC RISK ASSESSMENT REVIEW GROUP ON THE REACTOR SAFETY STUDY
E 1979
E U
D 3 PPS, 1 REF, NUCLEAR SAFETY, 20(1), PP. 24-26 (JAN.-FEB. 1979)
EGORY 010000
TION 0088
NTRY A
TRACT THE 7 MEMBER INDEPENDENT ASSESSMENT GROUP, HEADED BY DR. HAROLD LEWIS, WAS APPOINTED BY NRC IN 1977 TO CLARIFY THE BENEFITS AND LIMITATIONS OF THE RASMUSSEN REPORT, AND TO ASSESS COMMENTS THAT HAD BEEN MADE ABOUT IT. THE REPORT CONTAINS NUMEROUS FINDINGS AND RECOMMENDATIONS THAT NRC IS NOW CONSIDERING. (FAH)
WORDS PROBABILITY;ACCIDENT;STATISTICAL ANALYSIS;LICENSING PROCESS; AGENCY, NRC;SAFETY REVIEW;RISK

SSION NO. 00X0169826
LE PROBABILISTIC APPLICATION OF FRACTURE MECHANICS (IN ENGLISH)
HOR(S) DUFRESNE J
PAUTH CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
E 1981
E N
O DSN 451E + FRRSR-297. 15 PPS, 4 TABS, 24 REFS, APRIL 1981
IL AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
EGORY 230000;110000
TION 0135
NTRY F
TRACT THE DIFFERENT METHODS USED TO EVALUATE THE RUPTURE PROBABILITY OF A PRESSURE VESSEL ARE REVIEWED. DATA COLLECTION AND PROCESSING OF ALL PARAMETERS NECESSARY FOR FRACTURE MECHANICS EVALUATION ARE PRESENTED WITH PARTICULAR ATTENTION TO THE SIZE DISTRIBUTION OF DEFECTS IN ACTUAL VESSELS. PHYSICAL PROCESS IS FOLLOWED DURING CRACK GROWTH AND UNSTABLE PROPAGATION, USING LEFM AND PLASTIC INSTABILITY. RESULTS SHOW THAT THE FINAL FAILURE PROBABILITY FOR A PWR PRESSURE VESSEL IS 3.5 10(-6), AND IS DUE ESSENTIALLY TO LOCAS FOR ANY BREAK SIZE. THE WEAKEST POINT IS THE INTERNAL SIDE OF THE BELT LINE. (FAH)
WORDS FRANCE;FRACTURE TOUGHNESS;PRESSURE VESSELS;FLAW;CRACK;TEST, NONDESTRUCTIVE;FOREIGN EXCHANGE;PROBABILISTIC RISK ASSESSMENT; RISK;FAILURE, PRESSURE VESSEL;REACTOR, PWR

)/0000001-00001157/ 2
SSION NO. 00X0169788
LE FIRE RISK ANALYSIS FOR NUCLEAR POWER PLANTS
HOR(S) APOSTOLAKIS G;KAZARIANS M
PAUTH UNIV. OF CALIF., LOS ANGELES
E 1981
E N
O NUREG/CR-2258 + UCLA-ENG-8102 +. 185 PPS, 14 TABS, 13 FIGS, REFS, SEPT. 1981
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 170000;120000;230000
TION 0135
CODE UAV
NTRY A
TRACT A METHODOLOGY FOR EVALUATING THE FREQUENCY OF SEVERE CONSEQUENCES DUE TO FIRES IN NUCLEAR POWER PLANTS IS PRESENTED. THE METHODOLOGY PRODUCES A LIST OF ACCIDENT SCENARIOS AND THEN ASSESSES THE FREQUENCY OF OCCURRENCE OF EACH. ITS FRAMEWORK IS GIVEN IN SIX STEPS. IN THE FIRST TWO STEPS, THE ACCIDENT SCENARIOS ARE IDENTIFIED QUALITATIVELY AND THE POTENTIAL OF FIRES TO CAUSE INITIATING EVENTS IS INVESTIGATED. THE FREQUENCY OF FIRES IS OBTAINED FOR DIFFERENT COMPARTMENTS IN NUCLEAR POWER PLANTS USING BAYESIAN TECHNIQUES. THE RESULTS ARE COMPARED WITH THOSE OF CLASSICAL METHODS AND THE VARIATION OF THE FREQUENCIES WITH TIME IS ALSO EXAIMINED. THE COMBINED EFFECTS OF FIRE GROWTH, DETECTION, AND SUPPRESSION ON COMPONENT FAILURE ARE MODELED. THE SUSCEPTIBILITY OF CABLES TO FIRE AND THEIR FAILURE MODES ARE DISCUSSED.
ORDS FIRE PROTECTION;FIRE;RISK;ANALYTICAL MODEL;ANALYTICAL TECHNIQUE; CABLES AND CONNECTORS;ACCIDENT ANALYSIS;STATISTICAL ANALYSIS; REACTOR, PWR;SMOKE;INSTRUMENT, ALARM;PROBABILISTIC RISK ASSESSMENT

)/0000001-00001157/ 3
SSION NO. 00E0169773
E REACTOR SAFETY STUDY METHODOLOGY APPLICATIONS PROGRAM: GRAND GULF #1 BWR POWER PLANT
OR(S) HATCH SW;CYBULSKIS P;WOOTON RO
UTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM ; BATTELLE COLUMBUS LABS., OH
1981
H
NUREG/CR-1659 (4 OF 4) + SAND80-1897 (4 OF 4) +. APPROX. 500 PPS, FIGS, REFS, OCT. 1981
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

EGORY 050000;230000
TION 0135
P CODE AUA;CAF
NTRY A
TRACT THIS REPORT IS THE FOURTH IN A SERIES OF FOUR REPORTS WHICH PRESENT THE RESULTS OF ANALYSES PERFORMED IN THE REACTOR SAFETY STUDY METHODOLOGY APPLICATIONS PROGRAM (RSSMAP). THIS VOLUME DESCRIBES THE ANALYSIS PERFORMED FOR THE GRAND GULF UNIT 1 NUCLEAR REACTOR; OTHER VOLUMES DESCRIBE THE ANALYSES OF SEQUOYAH UNIT 1, OCONEE UNIT 3, AND CALVERT CLIFFS UNIT 2. LOSS OF COOLANT ACCIDENTS (LOCAS) AND TRANSIENTS WERE USED AS INITIATING EVENTS. THE MOST SIGNIFICANT SEQUENCES CONTRIBUTING TO THE CORE MELT FREQUENCY AND, BY EXTENSION, THE RISK WERE TRANSIENT INITIATED SEQUENCES WHICH ARE FOLLOWED BY A LOSS OF ALL LONG-TERM DECAY HEAT REMOVAL. THESE SEQUENCES CONTRIBUTED APPROXIMATELY 90% OF THE TOTAL CORE MELT FREQUENCY AT GRAND GULF. (FAH)
WORDS SAFETY ANALYSIS;ACCIDENT, LOSS OF COOLANT;CORE MELTDOWN;RISK; DECAY HEAT;REACTOR, BWR;GRAND GULF 1 (BWR);REACTOR TRANSIENT

/0/0000001-00001157/ 4
ESSION NO. 00J0169748
LE AN EVALUATION OF CONTAINMENT INERTING AND AIR DILUTION SYSTEMS AS METHODS FOR POST-ACCIDENT HYDROGEN CONTROL IN BWRS
HOR(S) HEISING-GOODMAN CD;LEPERVANCHE J
PAUTH MASS. INST. OF TECHNOLOGY, CAMBRIDGE, MA
E 1981
E O
G 18 PPS, 7 TABS, 8 FIGS, 25 REFS, NUCLEAR ENGINEERING & DESIGN. 64(3), PP. 329-46 (APRIL 1981)
EGORY 120000;230000
TION 0135
P CODE MSM
NTRY Z
BB NEDE
TRACT POST-ACCIDENT HYDROGEN GENERATION IN BWR CONTAINMENTS IS ANALYZED AS A FUNCTION OF ENGINEERED HYDROGEN CONTROL SYSTEM, ASSUMING EITHER NITROGEN INERTING OR AIR DILUTION. FAULT TREE ANALYSIS WAS APPLIED TO ASSESS THE FAILURE PROBABILITY PER DEMAND OF EACH SYSTEM. THESE FAILURE RATES WERE THEN COMBINED WITH THE PROBABILITY OF ACCIDENTS PRODUCING VARIOUS HYDROGEN GENERATION RATES TO CALCULATE THE OVERALL SYSTEM HYDROGEN CONTROL PROBABILITY. RESULTS INDICATE THAT BOTH SYSTEMS RENDER APPROXIMATELY THE SAME OVERALL SYSTEM HYDROGEN CONTROL FAILURE RATE (AIR DILUTION: 8.3 X 10(-2) -1.1 X 10(-2); NITROGEN INERTING: 1.5 X 10(-2) - 2 X 10(-3).
WORDS CONTAINMENT ATMOSPHERE;HYDROGEN;NITROGEN;FAULT TREE ANALYSIS; DILUTION;AIR;RISK;REACTOR, BWR;CORROSION;FIRE;EXPLOSION

/0/0000001-00001157/ 5
ESSION NO. 00X0169509
LE COMMENTS ON THE ACRS QUANTITATIVE SAFETY GOALS
HOR(S) STRIP DR
PAUTH SANDIA LABS., ALBUQUERQUE, NM
E 1981
E N
G NUREG/CR-2065 + SAND81-0899 +. 14 PPS, 1 FIG, 4 REFS, JUNE 1981
L AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 180000;230000
TION 0135
P CODE AUA
NTRY A
TRACT ACRS IN RESPONSE TO A REQUEST FROM THE NRC, PRESENTED A REPORT OUTLINING THE CONCEPTS OF QUANTITATIVE RISK ASSESSMENT CRITERIA AND MAKING RECOMMENDATIONS FOR SPECIFIC SAFETY GOALS. GOALS ARE SET FOR THE PROBABILITY OF REACTOR HAZARD STATES, RISK TO AN INDIVIDUAL, RISK TO SOCIETY, AND SOCIETAL IMPACT REDUCTION. DATA REGARDING ACCIDENT CONSEQUENCES AND THE INTERACTIONS OF THE CRITERIA, LEAD TO SOME CONCLUSIONS WHICH WERE NOT OBVIOUS. THE RISKS APPEAR SO MINIMAL THAT THE RISK CRITERIA PROPOSED IN THE ACRS REPORT ALLOW SITING IN APPARENT CONFLICT WITH EXISTING AND PROPOSED SITING POLICY. WHILE THE ACRS RECOMMENDATIONS

SEEM TO PROVIDE A USEFUL FRAMEWORK FOR A MORE ACCEPTABLE LICENSING PROCEDURE, SOME OF THE DETAILS MAY HAVE TO BE CHANGED TO RESOLVE CONFLICTS WITH NATIONAL SITING POLICIES. (FAH)

WORDS ACRS;RISK;CODES AND STANDARDS;POWER PLANT, NUCLEAR;SITING; ACCIDENT, CONSEQUENCES

/0/0000001-0000115// 6

ESSION NO. 0000169495

LE A STUDY OF THE IMPLICATIONS OF APPLYING QUANTITATIVE RISK CRITERIA IN LICENSING OF NUCLEAR POWER PLANTS IN THE UNITED STATES

HOR(S) MITRA S;HALL R;COPPOLA A

PAUTH BROOKHAVEN NATIONAL LAB., UPTON, NY

E 1981

E U

D NUREG/CR-2040 + BNL-NUREG-51367 +. 195 PPS. TABS. FIGS. REFS. MAY 1981

IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

EGORY 230000;180000

TION 0135

P CODE B2A

NTRY A

TRACT THIS REPORT DESCRIBES THE RESULTS OF AN INVESTIGATION INTO THE FEASIBILITY OF DEVELOPING AND USING A SET OF PROBABILISTIC RISK CRITERIA TO HELP JUDGE THE SAFETY OF NUCLEAR POWER PLANTS. THE HISTORICAL PROCESS WHICH HAS LED TO A DEVICE FOR NUMERICAL CRITERIA FOR REACTOR SAFETY IS REVIEWED. THE MCA (MAXIMUM CREDIBLE ACCIDENT) AND THE DBA (DESIGN BASIS ACCIDENT) CAPABILITIES AND LIMITATIONS ARE EXAMINED AND SEVERAL NEW CONCEPTS SUCH AS AVAILABILITY CRITERIA; SYSTEM AVAILABILITY CRITERIA; ACCIDENT PROBABILITY CRITERIA; RELEASE CRITERIA; INDIVIDUAL RISK AND SOCIAL RISK CRITERIA, ETC ARE PROPOSED AND DISCUSSED. 193 PAGES, 32 FIGURES, 10 TABLES AND NUMEROUS REFERENCES.

WORDS ACCIDENT, CONSEQUENCES;ACCIDENT, DESIGN BASIS;ACCIDENT ANALYSIS; SOCIO/PHILOSOPHICAL CONSIDERATION;SAFETY ANALYSIS;RISK;BENEFIT VS RISK;HAZARDS ANALYSIS;HAZARD, RELATIVE;ACCIDENT, MAXIMUM CREDIBLE (MCA);SAFETY ANALYSIS REPORT, METEOROLOGY;N-POWER, SAFETY OF;LICENSING PROCESS;PROBABILISTIC RISK ASSESSMENT

/0/0000001-0000115// 7

ESSION NO. 00X0169477

LE BWR OFF-GAS SYSTEMS-OPERATING EXPERIENCE AND PLANNING STUDY

HOR(S) NEGIN CA;WORKU G;KENWORTHY LD

PAUTH INTERNATIONAL ENERGY ASSOCIATES LTD., WASHINGTON, D.C.

E 1981

E N

D EPRI-NP-1839 +. 171 PPS, FIGS, MAY 1981

L AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10090, PALO ALTO, CALIF. 94303

EGORY 170000;230000;070000

TION 0135

NTRY A

TRACT THIS REPORT ADDRESSES BWR OFF-GAS SYSTEMS IN THE CONTEXT OF OPERATING PROBLEMS, PARTICULARLY WITH AUGMENTED SYSTEMS. SPECIAL EMPHASIS HAS BEEN PLACED ON EXPLOSIONS AND INTERNAL BURNING BECAUSE THESE EVENTS ARE OF THE GREATEST CONCERN AND HAVE THE MOST SIGNIFICANT IMPACT. PREVENTIVE MEASURES AND SAFETY PRECAUTIONS THAT APPLY TO EXPLOSIONS ARE DESCRIBED. MORE CONVENTIONAL TYPES OF OPERATING PROBLEMS ARE DESCRIBED, AND CORRECTIVE MEASURES ARE NOTED. CANDIDATE R & D PROJECTS ARE SUGGESTED. THIS REPORT CAN BE USED AS A REVIEW GUIDE AND FOR OFF-GAS SYSTEM OPERATING TRAINING. THE PRIMARY CONCLUSION IS THAT THE PROBABILITY OF AND RISKS FROM EXPLOSIONS HAVE DECREASED SIGNIFICANTLY AND WILL BE MINIMAL IN THE FUTURE. (FAH)

WORDS EPRI;REACTOR, BWR;OFF GAS;EXPLOSION;FIRE;OPERATING EXPERIENCE;R AND D PROGRAM;LICENSED OPERATOR;TRAINING;RISK

0/0000001-0000115// 8

SSION NO. 0000169002

E NUCLEAR ONE OF MANY MINOR RISKS

1980

L
I PG. NUCLEAR ENGINEERING INTERNATIONAL. 25(303). PG. 5 (SEPT. 1980)
EGORY 010000;230000
TION 0134
UNTRY A
STRACT UNITED STATES: LIVING NEAR A NUCLEAR POWER PLANT FOR 50 YEARS IS AS DANGEROUS AS SPENDING TWO DAYS IN NEW YORK OR TRAVELING IN A CANOE FOR SIX MINUTES, SAYS PROFESSOR RICHARD WILSON OF HARVARD UNIVERSITY. OTHER ACTIVITIES WHICH INCREASE AN INDIVIDUAL'S CHANCE OF DEATH BY ONE PART IN A MILLION ARE: SMOKING 1.4 CIGARETTS; EATING 40 TABLESPOONS OF PEANUT BUTTER; DRINKING MIAMI CITY WATER FOR ONE YEAR; EATING 100 CHARCOAL-BROILED STEAKS; SPENDING ONE HOUR IN A COAL MINE OR TWO MONTHS IN DENVER OR TWO MONTHS IN A STONE HOUSE; LIVING 20 YEARS NEAR A POLYVINYLCHLORIDE PLANT; DRINKING 1000 24 OZ. SOFT DRINKS FROM PLASTIC BOTTLES. THE RISK OF LIVING NEAR A NUCLEAR PLANT IS LOW ENOUGH TO BE JUDGED AS ONE OF THE MANY MINOR RISKS. (FAH)
YWORDS RISK;POWER PLANT, NUCLEAR;LOW;EXPOSURE, BACKGROUND;COMPARISON

//0/0000001-0000115// 9
CESSION NO. 00C0168969
TLE THE EXPANDING ROLE OF QUANTITATIVE RISK ANALYSIS IN THE FEDERAL REPUBLIC OF GERMANY
THOR(S) BIRKHOFER A
RPAUTH TECHNISCHE UNIVERSITAT MUNCHEN, F.R. GERMANY
TE 1981
PE L
D STI/PUB/566 (VOL. I) +. 13 PPS, PP. 307-19 OF PROCEEDINGS OF AN INTERNATIONAL CONFERENCE ON CURRENT NUCLEAR POWER PLANT SAFETY ISSUES; STOCKHOLM, SWEDEN, OCT. 20-24, 1980
IL AVAILABILITY - UNIPUB, INC., P.O. BOX 433, NEW YORK, N.Y. 10016
EGORY 230000
TION 0134
NTRY W
TRACT A DETAILED RISK STUDY HAS BEEN PERFORMED FOR A NUCLEAR POWER PLANT WITH A PWR. THIS WORK WAS NOT PART OF THE LICENSING PROCEDURE. ITS MAIN TASK WAS TO GIVE A FIRST ASSESSMENT OF THE ACCIDENTAL RISK FROM NUCLEAR POWER PLANTS IN THE FEDERAL REPUBLIC OF GERMANY, AND TO HELP TO GAIN EXPERIENCE WITH THE APPLICATION OF RISK ASSESSMENT METHODOLOGY. NEVERTHELESS, INSIGHT GAINED FROM THE STUDY IS INFLUENCING ACTUAL LICENSING PROCEDURES. THE IMPORTANCE OF RISK ANALYSIS SEEMS TO BE INCREASING, SUPPORTING NOT ONLY TECHNICAL, BUT ALSO LEGAL AND POLITICAL DECISIONS MAINLY BY SUPPLYING BACKGROUND INFORMATION.
WORDS GERMANY;RISK,RELIABILITY ANALYSIS;REACTOR, PWR;CORE MELTDOWN

//0/0000001-0000115// 10
ESSION NO. 00X0168860
LE AN INFORMATION SYSTEM FOR THE COLLECTION AND EVALUATION OF RELIABILITY DATA AT THE NUCLEAR POWER PLANT BIBLIS B
HOR(S) SCHWAIGER R;KEINLSCHMIDT G
PAUTH GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY
E 1981
E N
GRS-A-560 + GERRSR-742 +. 109 PPS, FIGS, REFS, FEB. 1981
GUAGE GERMAN
IL AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
EGORY 170000;230000
TION 0133
P CODE GRS
NTRY G
TRACT THE AIM OF THE PROJECT IS TO OBTAIN REPRESENTATIVE RELIABILITY DATA ON COMPONENTS AND SYSTEMS FOR RISK AND RELIABILITY ANALYSIS DURING THE LICENSING PROCESS BY A DATA COLLECTION IN THE NUCLEAR POWER PLANT (NPP) BIBLIS B. THE FOUR MAIN OBJECTIVES OF THIS PROJECT ARE: A) DEVELOPMENT OF A COLLECTION SYSTEM, B) COLLECTION OF RAW DATA, C) DEVELOPMENT OF AN EDP-BASED INFORMATION SYSTEM, D) EVALUATION OF RELIABILITY DATA. THIS INFORMATION SYSTEM, OFFERS THE POSSIBILITY OF CALCULATING RELIABILITY DATA FROM THE AVAILABLE RAW DATA.

POSSIBLE METHODS FOR SUCH AN INFORMATION SYSTEM ARE DISCUSSED. SYSTEM 2000, WHICH IS BEING USED FOR THIS PROJECT, IS PRESENTED IN MORE DETAIL. (FAH)

...RDS GERMANY;DATA COLLECTION;DATA PROCESSING;RISK;RELIABILITY ANALYSIS;POWER PLANT, NUCLEAR;FOREIGN EXCHANGE;LICENSING PROCESS

/0000001-00001157/ 11

...SSION NO. 00E0168879
...E REACTOR SAFETY STUDY METHODOLOGY APPLICATIONS PROGRAM: OCONEE #3 POWER PLANT
...OR(S) KOLB GJ;HATCH SW;WOOTON RO
...AUTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM
1981
H
NUREG/CR-1659 (2 OF 4) (REV.) +. APPROX. 300 PPS, FIGS, MAY 1981
...L AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
...GORY 170000;230000
...ION 0133
... CODE AUA
...TRY A
...RACT THIS VOLUME REPRESENTS THE RESULTS OF THE ANALYSIS OF OCONEE UNIT 3 NUCLEAR POWER PLANT WHICH WAS PERFORMED AS PART OF THE REACTOR SAFETY STUDY METHODOLOGY APPLICATIONS PROGRAM (RSSMAP). THE RSSMAP WAS CONDUCTED TO APPLY THE METHODOLOGY DEVELOPED IN THE REACTOR SAFETY STUDY (RSS) TO AN ADDITIONAL GROUP OF PLANTS WITH THE FOLLOWING OBJECTIVES IN MIND: 1) IDENTIFICATION OF THE RISK DOMINATING ACCIDENT SEQUENCES FOR A BROADER GROUP OF REACTOR DESIGNS; 2) COMPARISON OF THESE ACCIDENT SEQUENCES WITH THOSE IDENTIFIED IN THE RSS; AND 3) BASED ON THIS COMPARISON, IDENTIFICATION OF DESIGN DIFFERENCES WHICH HAVE A SIGNIFICANT IMPACT ON RISK. (FAH)
...ORDS OCONEE 3 (PWR);SAFETY REVIEW;RISK;DESIGN;COMPARISON;HJCK;NRC-AN; FAILURE, SEQUENTIAL

/0000001-00001157/ 12

...SSION NO. 00X0168877
...E THE USE OF RISK AVERSION IN RISK ACCEPTANCE CRITERIA
...OR(S) SIMPSON M;OKRENT D;GRIESMEYER JM
...AUTH UNIV. OF CALIF., LOS ANGELES
1980
N
ALO-83 + UCLA-ENG-7970 +. 27 PPS, 11 FIGS, 16 REFS, JUNE 1980
...L AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
...GORY 230000
...ION 0133
... CODE UAV
...TRY A
...RACT QUANTITATIVE RISK ACCEPTANCE CRITERIA FOR TECHNOLOGICAL SYSTEMS MUST BE BOTH JUSTIFIABLE, BASED UPON SOCIETAL VALUES AND OBJECTIVES, AND WORKABLE IN THE SENSE THAT COMPLIANCE IS POSSIBLE AND CAN BE DEMONSTRATED IN A STRAIGHT FORWARD MANNER. SOCIETAL VALUES HAVE FREQUENTLY BEEN ASSESSED USING RECORDED ACCIDENT STATISTICS ASSUMING THAT THE STATISTICS REFLECT, IN SOME WAY, SOCIETAL PREFERENCES. IN THIS REPORT THE IMPLICATION OF INCORPORATING RISK AVERSION IN ACCEPTANCE CRITERIA IS INVESTIGATED. CALCULATED RISKS OF VARIOUS TECHNOLOGICAL SYSTEMS ARE CONVERTED TO EXPECTED SOCIAL COSTS. THE UNCERTAINTIES ARE DISCUSSED.
...ORDS SAFETY PRINCIPLES AND PHILOSOPHY;DECISION ANALYSIS;BENEFIT VS RISK;SOCIO/PHILOSOPHICAL CONSIDERATION;RISK

/0000001-00001157/ 13

...SSION NO. 00X0168423
...E POTENTIAL SAFETY-RELATED INCIDENTS WITH POSSIBLE APPLICABILITY TO A NUCLEAR FUEL REPROCESSING PLANT
...OR(S) DEXTER AH;DURANT WS;PERKINS WC
...AUTH SAVANNAH RIVER LAB., AIKEN, SC
1980
N
DP-1558 +. 304 PPS, FIGS, REFS, DEC. 1980

L AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S.
DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 130000;170000;230000
TION 0133
P CODE SRL
NTRY A
TRACT THE OCCURRENCE OF CERTAIN POTENTIAL EVENTS IN NUCLEAR FUEL
REPROCESSING PLANTS COULD LEAD TO SIGNIFICANT CONSEQUENCES
INVOLVING RISK TO OPERATING PERSONNEL OR TO THE GENERAL PUBLIC.
THIS DOCUMENT IS A COMPILATION OF SUCH POTENTIAL INITIATING
EVENTS IN NUCLEAR FUEL REPROCESSING PLANTS. POSSIBLE GENERAL
INCIDENTS AND INCIDENTS SPECIFIC TO KEY OPERATIONS IN FUEL
REPROCESSING ARE CONSIDERED, INCLUDING POSSIBLE CAUSES,
CONSEQUENCES, AND SAFETY FEATURES DESIGNED TO PREVENT, DETECT,
OR MITIGATE SUCH INCIDENTS.
WORDS FUEL REPROCESSING;OPERATING EXPERIENCE;INCIDENT;INCIDENT, HUMAN
ERROR;ACCIDENT;RISK;SAFETY ANALYSIS;CONTAMINATION;ENGINEERED
SAFETY FEATURE;FIRE;EXPLOSION;WASTE TREATMENT;ACCIDENT ANALYSIS;
FILTERS

/70/0000001-000011577 14
ESSION NO. 0000168420
LE TESTIMONY PRESENTED TO THE HOUSE SCIENCE AND TECHNOLOGY
COMMITTEE
HOR(S) RICHMOND CR
PAUTH OAK RIDGE NATIONAL LAB., TN
E 1981
E U
D ORNL/TM-8023 +. 30 PPS, 2 TABS, OCT. 1981
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S.
DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 230000;610000
TION 0133
P CODE F2C
NTRY A
TRACT THIS REPORT IS THE TEXT OF INVITED TESTIMONY GIVEN BY THE
AUTHOR BEFORE THE HOUSE SCIENCE AND TECHNOLOGY COMMITTEE. THIS
CONGRESSIONAL HEARING ON SOCIETAL RISKS OF ENERGY SYSTEMS
REFLECTS THE GROWING INTEREST ON THE PART OF CONGRESS, THE
PUBLIC, THE SCIENTIFIC COMMUNITY, AND OTHER GROUPS ON THIS
EXTREMELY IMPORTANT TOPIC OF RISK ANALYSIS. THIS PRESENTATION
CONTAINS INFORMATION ON THE EMERGENCE OF AN INTERDISCIPLINARY
PROFESSIONAL FIELD OF RISK ANALYSIS, INCLUDING THE RECENTLY
FORMED "SOCIETY FOR RISK ANALYSIS." THE RISK ANALYSIS PROGRAMS
PRESENTLY UNDERWAY AT ORNL ARE BRIEFLY DISCUSSED.
WORDS CONGRESSIONAL ACTIVITY;BENEFIT VS RISK;SOCIO/PHILOSOPHICAL
CONSIDERATION;RISK;HAZARD, RELATIVE;HAZARDS ANALYSIS

/0/0000001-000011577 15
ESSION NO. 00X0168371
LE RISKS OF NUCLEAR ENERGY & LOW-LEVEL IONIZING RADIATION
PAUTH AMERICAN MEDICAL ASSN., MONROE, WI
E 1981
E N
D 40 PPS, 1981
GUAGE OTHER LANG
IL AVAILABILITY - ORDER DEPT. OP-125, AMERICAN MEDICAL ASSN., P.O.
BOX 821, MONROE, WI 53566
EGORY 230000
TION 0133
NTRY E
TRACT THE AMERICAN MEDICAL ASSOCIATION (AMA)'S STAFF PREPARED AN
OVERVIEW REPORT BASED UPON CURRENT INFORMATION THAT GAVE A
RANGE OF ESTIMATES OF THE OCCUPATIONAL AND NON-OCCUPATIONAL
HEALTH EFFECTS OF COAL, OIL, NUCLEAR AND NATURAL GAS GENERATED
ELECTRICITY. NUCLEAR ENERGY AND NATURAL GAS WERE ESTIMATED TO
CAUSE THE FEWEST OCCUPATIONAL IMPAIRMENTS PER 1000 MEGAWATT
ELECTRICITY-GENERATING UNIT WITH COAL AND OIL HAVING THE
HIGHEST IMPAIRMENTS. TWO BROAD QUESTIONS ARE DISCUSSED; (1)
HOW MUCH ELECTRICITY DOES THE U.S. SOCIETY DEMAND AND (2) WHAT
TECHNOLOGIES SHOULD BE USED TO GENERATE THAT AMOUNT OF
ELECTRICITY? THIS REPORT FOCUSES ON THE HEALTH EFFECTS OF THE
SOURCES OF ELECTRIC POWER.
WORDS MORTALITY;RADIATION EXPOSURE;RISK;PERSONNEL EXPOSURE, RADIATION;

HAZARD, RELATIVE;RADIATION SAFETY AND CONTROL;N-POWER, SAFETY OF

0/0000001-00001157/   16
SSION NO.   00J0168355
LE   USING COMPARATIVE RISK ASSESSMENT
HOR(S)   GREENHOUSE G
E   1981
    U
    1 PG. NUCLEAR ENGINEERING INTERNATIONAL. 26(318), PG. 15 (SEPT. 1981)
EGORY   230000
TION   0133
NTRY   U
DD   NEIN
TRACT   COMPARING NUCLEAR RISK WITH THE RISK OF COAL BURNING HAS NOT BROUGHT ABOUT ANY INCREASE IN PUBLIC ACCEPTANCE OF NUCLEAR POWER. RATHER COMPARATIVE RISK ASSESSMENTS HAVE ONLY CONTRIBUTED CONSIDERABLY TO THE DIFFICULTIES IN LICENSING COAL-FIRED POWER STATIONS. THE SOMETIMES SENSATIONAL COVERAGE GIVEN TO COMPARATIVE RISK ASSESSMENT STUDIES HAS "ADDED TO PUBLIC FEARS, FRUSTRATION AND DISTRUST OF THE POWER INDUSTRY AS A WHOLE". RISK ASSESSMENT SHOULD GIVE FULL WEIGHT TO THE ECONOMIC, SOCIAL, AND HEALTH RISKS OF A FAILURE TO PROVIDE ADEQUATE SUPPLIES.
WORDS   GERMANY;RISK;BENEFIT VS RISK;FUEL, FOSSIL;COAL;ENERGY;UNITED KINGDOM

0/0000001-00001157/   17
ESSION NO.   00X0168347
LE   DETERMINATION OF REDUCTION IN LIFE EXPECTANCY FROM STOCHASTIC SOMATIC FATALITIES AFTER ACCIDENTAL RADIATION-EXPOSURE (IN GERMAN)
HOR(S)   EHRHARDT J
PAUTH   KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
E   1981
    N
    KFK-3181 + GRRSR-728 +. 115 PPS, 22 TABS, 7 FIGS, 6 REFS, JUNE 1981
GUAGE   GERMAN
IL   AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
EGORY   230000
TION   0133
P CODE   KSR
NTRY   G
TRACT   IN RISK STUDIES FOR PLANTS OF THE NUCLEAR INDUSTRY USUALLY THE NUMBER OF FATALITIES BY RADIATION INDUCED HEALTH EFFECTS ARE ASSESSED. THEY ARE DIVIDED INTO "EARLY FATALITIES" (MORTALITY BY ACUTE RADIATION SYNDROM) AND "LATE FATALITIES" (DEATHS DUE TO LEUKEMIA AND CANCER). FOR THE INDIVIDUALS AS WELL AS FOR THE SOCIETY AS A WHOLE BOTH HEALTH EFFECTS ARE IN PRINCIPLE OF DIFFERENT SIGNIFICANCE BECAUSE OF THE DIFFERENT TIMES OF DEATH AFTER IRRADIATION. RISK ASSESSMENTS WHICH GIVE ONLY THE NUMBER OF FATALITIES THEREFORE SHOW AN INCOMPLETE PICTURE OF THE CONSEQUENCE BECAUSE THEY DO NOT CONSIDER THE AGE OF THE INDIVIDUALS AT THE TIME OF DEATH. IN THIS REPORT THE MATHEMATICAL MODELS FOR THE COMPUTATION OF THE INDIVIDUAL AND COLLECTIVE REDUCTION OF LIFE EXPECTANCY FROM STOCHASTIC SOMATIC EFFECTS AFTER ACCIDENTAL RELEASES OF RADIOACTIVITY ARE DESCRIBED. COMPUTATIONAL RESULTS WITH REGARD TO THE AGE DISTRIBUTION OF THE POPULATION ARE PRESENTED FOR PERSONS LIVING DURING THE NUCLEAR ACCIDENT AND PERSONS BORN AFTERWARDS. THEREBY THE EXPOSURE PATHWAYS, ORGANS AND NUCLIDES OF THE GERMAN RISK STUDY ARE TREATED SEPARATELY. A RAW ESTIMATION OF THE REDUCTION IN LIFE EXPECTANCY DUE TO THE LATE FATALITIES CALCULATED IN THIS STUDY IS GIVEN.
ORDS   PERSONNEL EXPOSURE, RADIATION;RADIATION EXPOSURE;HAZARD, RELATIVE;RISK;MODEL, STOCHASTIC;MODEL, BIOLOGICAL;RADIATION MODEL;MORTALITY;FOREIGN EXCHANGE

0/0000001-00001157/   18
SSION NO.   00C0168278
E   NATIONAL ACADEMY OF SCIENCES SURVEY ON RISKS ASSOCIATED WITH

NUCLEAR POWER
HOR(S) BUCHANAN JR
PAUTH OAK RIDGE NATIONAL LAB., TN
E 1980
E L
O 8 PPS. FROM 1980 ANS TOPICAL MEETING ON THERMAL REACTOR SAFETY; KNOXVILLE, TN, APRIL 1980
AIL AVAILABILITY - J.R. BUCHANAN, OAK RIDGE NATIONAL LAB., P.O. BOX Y, OAK RIDGE, TN 37830
TEGORY 230000
ITION 0133
RP CODE F2C
UNTRY A
STRACT A CRITICAL REVIEW OF THE LITERATURE PERTAINING TO THE RISKS ASSOCIATED WITH NUCLEAR ELECTRIC POWER WAS SPONSORED BY THE COMMITTEE ON SCIENCE AND PUBLIC POLICY OF THE NATIONAL ACADEMY OF SCIENCES. ALTHOUGH THE FULL REPORT (CONSISTING OF OVER 25 CHAPTERS) HAS NOT YET BEEN PUBLISHED, THIS PAPER PRESENTS HIGHLIGHTS FROM THE "SUMMARY AND SYNTHESIS CHAPTER," WHICH WAS RELEASED SEPARATELY. OF THE RISKS WHOSE MAGNITUDES CAN BE ESTIMATED WITH REASONABLE ACCURACY, THE MOST SERIOUS IS THE EXPOSURE OF FUTURE GENERATIONS TO (14)C FROM REACTORS AND REPROCESSING PLANTS. PROSPECTS ARE GOOD FOR REDUCING THIS RISK CONSIDERABLY, SINCE CARBON CAN BE COLLECTED AND STORED AS WASTE.
YWORDS RISK;REVIEW;CARBON;RADIOISOTOPE;MINING;URANIUM;THORIUM;FUEL CYCLE;FUEL REPROCESSING;FIRE;SABOTAGE;DOSE;RADON;RELIABILITY ANALYSIS;WASTE DISPOSAL;WASTE MANAGEMENT;THEFT/DIVERSION

7/07000001-00001157/ 19
CESSION NO. 00C0168271
LE NUCLEAR SAFETY IN PERSPECTIVE
HOR(S) SIDDALL E
PAUTH CANATOM LTD.
E 1979
E L
O 35 PPS. PRESENTED AT THE CANADIAN NUCLEAR ASSOCIATION CONVENTION; TORONTO, CANADA, JUNE 10-13, 1979
EGORY 230000;010000
ITION 0133
UNTRY G
STRACT A METHOD IS PROPOSED WHICH WOULD POINT TO AN OPTIMUM EXPENDITURE ON NUCLEAR SAFETY MEASURES. A DEPARTURE FROM THIS OPTIMUM IN EITHER DIRECTION WOULD RESULT IN A NET OVERALL LOSS OF SAFETY. THE FIRST FACTOR IN THE OPTIMIZATION IS THE RELATIONSHIP BETWEEN LIVES SAVED AND TOTAL ADDED COST OF SAFETY MEASURES AND PROCEDURES IN THE NUCLEAR INDUSTRY. THE COUNTERVAILING FACTOR IS THE MARGINAL COST OF PREVENTING OR AVOIDING PREMATURE DEATHS IN THE COMMUNITY. THE ANALYSIS INDICATES THAT PRESENT EXPENDITURES ON NUCLEAR SAFETY ARE FAR IN EXCESS OF THE OPTIMUM.
WORDS CANADA;POLLUTION IN PERSPECTIVE;RADIATION IN PERSPECTIVE;RISK;OPTIMIZATION;MORTALITY;INDUSTRY, NUCLEAR;POWER PLANT, FOSSIL FUEL;SCRUBBER;COST BENEFIT

7/07000001-00001157/ 20
ESSION NO. 00J0167894
LE TECHNICAL NOTE: PERSPECTIVE AND LIMITATIONS OF COMPARATIVE RISK-BENEFIT-COST EFFECTIVENESS
HOR(S) VINCK W
PAUTH COMMISSION OF THE EUROPEAN COMMUNITIES, BRUSSELS, BELGIUM
E 1981
E J
O 4 PPS. NUCLEAR SAFETY, 22(5), PP. 594-97 (SEPT-OCT. 1981)
EGORY 230000
ITION 0132
NTRY A
B NUSA
TRACT QUANTIFICATION IN RISK ASSESSMENT AND IN COST-RISK-BENEFIT ASSESSMENT CONSTITUTE A SIGNIFICANT AID IN APPRECIATION OF WHAT IS "SAFE-ENOUGH". SCIENTIFIC APPROACHES IN ASSESSING SOCIETAL BEHAVIOR OF INDIVIDUALS AND/OR COMMUNITIES WITH REGARD TO ACCEPTANCE INDICATE THAT QUANTIFICATION MAY BE LESS RELIABLE THAN COST-RISK-BENEFIT ASSESSMENTS. THE APPROACH IN THIS NOTE

IS TO SEEKING A SYSTEMATIC RATHER THAN INTUITIVE ACCEPTABILITY OF MAN-CREATED RISKS.
WORDS RISK;BENEFIT VS RISK;ANALYTICAL MODEL;SOCIO/PHILOSOPHICAL CONSIDERATION;HAZARD, RELATIVE;HAZARDS ANALYSIS;ECONOMIC STUDY; COST BENEFIT

/7/0/0000061-00001157/ 21
CESSION NO. 00J0167893
TLE EQUITY ASPECTS OF RISK MANAGEMENT: TRADE-OFFS BETWEEN PUBLIC AND OCCUPATIONAL HAZARDS IN NUCLEAR INDUSTRY
THOR(S) LOMBARD J;FAGNANI F
RPAUTH CEA CENTRE D'ETUDE SUR L'EVALUATION DE LA PROTECTION DANS LE DOMAINE NUCLEAIRE, FRANCE
TE 1981
PE 0
D 7 PPS. 7 TABS. 10 REFS. NUCLEAR SAFETY, 22(5), PP. 570-76 (SEPT.-OCT. 1981)
TEGORY 230000
ITION 0132
UNTRY A
ABS NUSA
STRACT EQUITY PROBLEMS ARISE IN RISK MANAGEMENT WHENEVER A SITUATION JUSTIFIES DIVIDING THE EXPOSED POPULATION INTO SPECIFIC SUBGROUPS. THE USUAL PRACTICE IN RADIATION PROTECTION IN THE NUCLEAR FUEL CYCLE, WHICH CONSISTS ROUGHLY IN DIVIDING THE POPULATION INTO TWO GROUPS (PLANT EMPLOYEES AND THE GENERAL PUBLIC), ASSUMES THAT MANAGEMENT OF THE PROTECTION SYSTEM FOR EACH GROUP IS RELATIVELY INDEPENDENT. BUT IN THE CASE OF PUBLIC PROTECTION AGAINST RISKS ASSOCIATED WITH PRESSURIZED-WATER REACTORS, FOR EXAMPLE, IT IS SHOWN THAT THIS HYPOTHESIS OF INDEPENDENCE IS NOT VALID AND THAT THERE ARE SIGNIFICANT RISK TRADE-OFFS BETWEEN THE PUBLIC AND THE EMPLOYEES. THIS LEADS ONE TO QUESTION THE VALUATION OF THE IMPLICIT VALUE OF A MAN-SIEVERT, DEPENDING ON WHETHER ONE IS DEALING WITH EMPLOYEES OR THE PUBLIC, FOR THE ANSWER TO THIS QUESTION STRONGLY AFFECTS THE RESULT OF ANY OPTIMIZATION OF CHOICES IN RADIATION PROTECTION.
WORDS RISK;BENEFIT VS RISK;HAZARDS ANALYSIS;ACCIDENT ANALYSIS;SAFETY ANALYSIS;SYSTEM ANALYSIS;SOCIO/PHILOSOPHICAL CONSIDERATION

/0/0000001-00001157/ 22
ESSION NO. 00J0167879
LE VERIFICATION OF FAULT TREE ANALYSIS VOL. 2: TECHNICAL DESCRIPTION
HOR(S) ROTHBART G;NEWT J;BASIN S
PAUTH SCIENCE APPLICATIONS INC., PALO ALTO, CA
E 1981
E U
EPRI-NP-1570 (VOL. 2) +. 150 PPS. 15 TABS. 25 FIGS. MAY 1981
IL AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10090, PALO ALTO, CALIF. 94303
EGORY 230000
TION 0132
P CODE SAI
NTRY A
TRACT AN ELECTRONIC INSTRUMENT HAS BEEN DEVELOPED TO SIMULATE THE RELIABILITY OF COMPLEX SAFETY SYSTEMS. USING DIGITAL INTEGRATED CIRCUITS ON MODULAR PRINTED CIRCUIT BOARDS, TOGETHER WITH A MONITORING MICROCOMPUTER SYSTEM AND OTHER SUPPORT HARDWARE, IT IS POSSIBLE TO SIMULATE SYSTEMS COMPOSED OF UP TO TWENTY INDEPENDENT COMPONENTS TEN BILLION TIMES FASTER THAN REAL-TIME. ARBITRARY TIME-DEPENDENT HAZARD FUNCTIONS, COMPLEX REPAIR MECHANISMS AND PROCEDURES, AND COMMON MODE INTERACTIONS ARE INCORPORATED INTO THE SYSTEM HARDWARE. THIS INSTRUMENT, TERMED ERMA (EPRI RELIABILITY AND MAINTAINABILITY ANALYZER), IS DESCRIBED IN DETAIL IN THIS REPORT WHICH CONTAINS THE DETAILS OF THE ELECTRONIC CIRCUITRY AND SUPPORTING SOFTWARE. A COMPANION, VOLUME I, DESCRIBES THE THEORY AND THE RESULTS OF EXPERIMENTS PERFORMED WITH ERMA.
ORDS ACCIDENT ANALYSIS;ACCIDENT MODEL;FAULT TREE ANALYSIS;FAILURE, COMMON MODE;SAFETY ANALYSIS;BENEFIT VS RISK;RISK;COMPUTER, DIGITAL;HAZARDS ANALYSIS

SSION NO. 00J0167674
LE NRC-INDUSTRY COOPERATIVE EFFORT SETS STAGE FOR PROBABILISTIC RISK ASSESSMENT STUDIES
AUTH EPRI NUCLEAR SAFETY ANALYSIS CENTER, PALO ALTO, CA
E 1981
E U
D 6 PPS. NUCLEAR NEWS, 24(11), PP. 87-92 (SEPT. 1981)
EGORY 230000
TION 0132
P CODE EPR
NTRY A
D NONE
TRACT THE U.S. NUCLEAR REGULATORY COMMISSION HAS JOINED WITH KEY INDUSTRY PERSONNEL TO DEVELOP PROCEDURE GUIDES TO BE USED FOR PROBABILISTIC RISK ASSESSMENTS. THESE PROCEDURES DOCUMENT THE STATE OF THE THE ART FOR ALL FACETS OF PROBABILISTIC RISK ANALYSIS AS APPLIED TO NUCLEAR POWER PLANTS. INCLUDED ARE DATA DEVELOPMENT, PLANT LOGIC MODELS, ACCIDENT SEQUENCE, IDENTIFICATION AND QUALIFICATION, ASSESSMENT OF PHYSICAL PROCESSES IN SEVEN ACCIDENTS, FISSION PRODUCT BEHAVIOR AND THE INTEGRATION OF THESE ITEMS TO FORM TOTAL RISK ASSESSMENT. THIS ARTICLE DISCUSSES THE PROJECT AND TESTS THE PROCEDURES.
WORDS ACCIDENT MODEL;ACCIDENT, CONSEQUENCES;RISK;SAFETY EVALUATION; MODEL, DETERMINISTIC;ACCIDENT, MAXIMUM CREDIBLE (MCA);HAZARDS ANALYSIS

07000001-00011577 24
SSION NO. 00J0167780
LE NUCLEAR POWER AND RADIATION IN PERSPECTIVE - SELECTIONS FROM NUCLEAR SAFETY
HOR(S) BUCHANAN JR;HARIED JA
AUTH OAK RIDGE NATIONAL LAB., TN
E 1981
E U
D NUREG/CR-1788 + ORNL/NUREG/NSIC-161 +. 321 PPS, MAY 1981
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 010000;230000;150000
TION 0131
P CODE F2C
NTRY A
TRACT THIS REVIEW COMPILES 33 ARTICLES ABOUT NUCLEAR POWER AND ASSOCIATED RADIATION HAZARDS WRITTEN FOR NUCLEAR SAFETY BETWEEN 1964 AND 1980. A PERSPECTIVE ON THESE HAZARDS IS SOUGHT BY COMPARING THEM OVER THESE LAST 16 YEARS WITH HAZARDS INHERENT IN OTHER ENERGY DEVELOPMENT TECHNOLOGIES. FOUR APPROACHES TO THE PROBLEM ARE CONSIDERED: BIOLOGICAL EFFECTS OF LOW-LEVEL RADIATION, RISK-BENEFIT CONCEPTS, NUCLEAR FUEL CYCLE RISKS AS COMPARED WITH OTHER RISKS, AND THE RELATIONSHIP BETWEEN MASS MEDIA AND PUBLIC INTEREST.
WORDS SOCIO/PHILOSOPHICAL CONSIDERATION;ENERGY POLICY;ENERGY SOURCE; COMPARISON;RISK;PUBLIC RELATIONS;POWER GENERATION METHOD; ELECTRIC POWER, ALTERNATE;RADIATION EFFECT;BIOMEDICAL

07000001-00011577 25
SSION NO. 00V0167426
LE RESPONSE TO INQUIRY CONCERNING THE SAFETY IMPLICATIONS OF CONTROL SYSTEMS FAILURES
AUTH U.S. NUCLEAR REGULATORY COMMISSION
E 1981
E M
D NRC NEWS RELEASE 81-78 +. 2 PPS, FOR WEEK ENDING MAY 19, 1981
IL AVAILABILITY - NRC, OFFICE OF PUBLIC AFFAIRS, WASHINGTON, D.C. 20555
EGORY 170000;230000
TION 0131
P CODE NRC
NTRY A
TRACT THE ACRS RECOMMENDED, IN A LETTER DATED AUG. 12, 1980, TO THE NRC THAT CONTROL SYSTEM RELIABILITY BE ADDED TO THE LIST OF UNRESOLVED SAFETY ISSUES BEING COMPILED BY THE NRC STAFF. THE NRC STAFF SUBSEQUENTLY ADDED TO ITS LIST OF UNRESOLVED SAFETY ISSUES AN ITEM DESIGNATED "SAFETY IMPLICATIONS OF CONTROL SYSTEMS." IN THE STAFF'S DESCRIPTION OF THIS ISSUE, EMPHASIS

WAS ON A STUDY OF CONTROL SYSTEM FAILURES THAT MIGHT DISABLE SAFETY SYSTEMS.
WORDS CONTROL SYSTEM;CONTROL ROD INTERACTION;SAFETY DEVICE;ACRS;RISK; FAILURE, EQUIPMENT

/07/0000001-00001157/ 26
ESSION NO. 00C0167388
LE COMPILATION OF THE PRESENTATION AT THE 1980 PROJECT NUCLEAR SAFETY COLLOQUIM, ATOMIC RESEARCH CENTER, KARLSRUHE, FRG, NOVEMBER 25, 1980 (IN GERMAN)
PAUTH KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
E 1981
E L
O KFK-3070 + GRRSR-709 +. VP, FEB. 1981
GUAGE GERMAN
IL AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
EGORY 180000;110000;050000;230000
TION 0131
P CODE GRS
NTRY G
TRACT TITLES OF THE PAPERS GIVEN ARE: 1) HUMAN FAILURE: A BASIC PROBLEM OF NUCLEAR POWER PLANTS, 2) RECENT ADVANCES OF REACTOR SAFETY RESEARCH IN THE NUCLEAR SAFETY PROJECT, 3) NRC'S CORE MELT RESEARCH PROGRAM AND ITS RELATION TO CURRENT REGULATORY ACTIVITIES, 4) U.S. STEAM EXPLOSION RESEARCH: RISK PERSPECTIVE AND EXPERIMENTAL RESULTS, 5) ANALYSIS OF THE COURSE OF HYPOTHETICAL CORE MELTDOWN ACCIDENTS, 6) TWO PHASE MASS FLOW MEASUREMENTS: COMPARISON OF DIFFERENT METHODS, 7) METHODS OF FLUID AND STRUCTURAL DYNAMICS APPLIED TO POSTULATED LWR ACCIDENTS, AND 8) MEASUREMENTS OF CLAD TEMPERATURES WITH LOFT-TYPICAL THERMOCOUPLES IN THE COSIMA FACILITY UNDER BLOWDOWN CONDITIONS.
WORDS HUMAN FACTORS;SAFETY PROGRAM;CORE MELTDOWN;EXPLOSION;STEAM;RISK; ACCIDENT, HYPOTHETICAL;FLOW, TWO PHASE;MEASUREMENT, TEMPERATURE; BLOWDOWN;GERMANY;FOREIGN EXCHANGE

/07/0000001-00001157/ 27
ESSION NO. 00C0167335
LE RISKS OF SHIPPING PLUTONIUM BY TRUCK, TRAIN, AND CARGO AIRCRAFT
HOR(S) JOHNSON JF;ANDREWS WB
PAUTH BATTELLE PACIFIC NORTHWEST LABS., RICHLAND, WA
E 1978
E L
O PNL-SA-6530 +. 6 PPS, 2 TABS, 4 FIGS, 9 REFS, MAY 1978
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 030000;220000
TION 0131
P CODE GNR
NTRY A
TRACT THE METHODOLOGY USED TO EVALUATE THE RISK IN SHIPPING PLUTONIUM IS DISCUSSED. THE METHODOLOGY IS COMPOSED OF FOUR BASIC STEPS: (1) A DETAILED DESCRIPTION OF THE TRANSPORTATION SYSTEM, (2) IDENTIFICATION OF POSSIBLE MATERIAL RELEASE SEQUENCES, (3) EVALUATION OF THE PROBABILITIES AND THE CONSEQUENCES OF THE RELEASES, AND (4) CALCULATION AND ASSESSMENT OF THE RISK.
WORDS TRANSPORTATION AND HANDLING;PLUTONIUM;ACCIDENT, PROBABILITY OF; ACCIDENT ANALYSIS;RADIATION EXPOSURE;BENEFIT VS RISK;RISK

/07/0000001-00001157/ 28
ESSION NO. 00C0167262
LE INSTITUTIONAL ISSUES AFFECTING TRANSPORTATION OF NUCLEAR MATERIALS
HOR(S) REESE RT;LUNA RE
PAUTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM
E 1980
E L
O SAND80-0902C + CONF-801115-26 +. 7 PPS, FROM 6TH INTERNATIONAL SYMPOSIUM ON PACKAGING & TRANSPORTATION OF RADIOACTIVE MATERIALS, NOV. 1980
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GGORY 020000;010000;230000
TION 0130
P CODE AUA
NTRY A
TRACT THE SHIPMENT OF NUCLEAR MATERIALS IN THE US NUCLEAR WASTE MANAGEMENT PROGRAM WILL REQUIRE THE SOLUTION OF BOTH TECHNICAL AND INSTITUTIONAL PROBLEMS. THE INDUSTRY HAS EXHIBITED ITS ABILITY TO SOLVE THE TECHNICAL PROBLEMS BUT THE INSTITUTIONAL PROBLEMS REQUIRE DIFFERENT APPROACHES, ASSESSMENTS AND EVALUATIONS. THE PAPER OUTLINES THE TRANSPORTATION TECHNOLOGY CENTERS (TTC) PROGRAM TO IDENTIFY THE INSTITUTIONAL ISSUES, THE METHODOLOGY DEVELOPED TO ASSESS AND EVALUATE WHAT EFFECTS ARE CREATED BY AN ISSUE AND WHAT, IF ANYTHING, CAN BE DONE ABOUT A PARTICULAR ISSUE.
WORDS TRANSPORTATION AND HANDLING;RISK;SOCIO/PHILOSOPHICAL CONSIDERATION;REGULATION, FEDERAL;REGULATION, STATE;TRAIN; TRAINING;WASTE TRANSPORTATION

/0/0000001-00001157/ 29
ESSION NO. 00J0167062
LE RISK ANALYSIS
E 1981
E O
O 95 PPS, RISK ANALYSIS, 1(1)(1981) (PUBLISHED BY PLENUM PRESS, N.Y.)
EGORY 230000
TION 0130
NTRY A
BB RIAN
TRACT THIS IS THE FIRST ISSUE OF THIS JOURNAL ON RISK ANALYSIS WHICH IS AN OFFICIAL PUBLICATION OF THE SOCIETY FOR RISK ANALYSIS. ARTICLES INCLUDED IN THIS ISSUE INCLUDE: ON THE QUANTITATIVE DEFINITION OF RISK; ENERGY PRODUCTION RISKS: WHAT PERSPECTIVE SHOULD WE TAKE?; REGULATION OF CARCINOGENS; ESTIMATED CANCER RISK ASSOCIATED WITH OCCUPATIONAL ASBESTOS EXPOSURE; RISK IN BENEFIT-COST ANALYSIS; AND VALUE OF A LIFE: WHAT DIFFERENCE DOES IT MAKE? VARIOUS COMMENTS ON THE ARTICLES AND LETTERS TO THE EDITOR ARE INCLUDED.
WORDS RISK;NUMERICAL METHOD;CARCINOGEN;CANCER;MORTALITY;REGULATION; COST BENEFIT;ENERGY POLICY;POWER PLANT, NUCLEAR;POWER PLANT, FOSSIL FUEL;COMPARISON

/0/0000001-00001157/ 30
ESSION NO. 00X0166902
LE RISKS REGULATION RESPONSIBILITIES AND COSTS IN NUCLEAR MANAGEMENT. A PRELIMINARY SURVEY IN THE EUROPEAN COMMUNITY
HOR(S) ORLOWSKI S;DAVID JL;ATTWATER NC
PAUTH COMMISSION OF THE EUROPEAN COMMUNITIES
E 1980
E N
O EUR-6893 EN, FR +. 37 PPS, 4 TABS, 19 REFS, 1980
TL AVAILABILITY - EUROPEAN COMMUNITY INFORMATION SERVICE, 2100 M ST., N.W., SUITE 707, WASHINGTON, D.C. 20027
EGORY 230000;010000
TION 0130
P CODE EAE
NTRY U
TRACT INDUSTRIALIZED SOCIETIES ARE AWARE OF THE NEED TO ESTABLISH SUITABLE CONTROLS TO PROTECT THE ENVIRONMENT AND THEIR CITIZENS FROM THE RISKS OF POLLUTION; CONTROLS MAY BE TECHNICAL OR INSTITUTIONAL. IDEALLY, THE DEVELOPMENT OF BOTH TYPES OF CONTROL SHOULD PROCEED IN PARALLEL SO THAT THE TECHNICAL PROCESSES COMPLY WITH THE SAFETY CRITERIA LAID DOWN IN THE INSTITUTIONAL CONTROLS AND, CONVERSELY, SO THAT THE ADMINISTRATIVE, LEGAL AND FINANCIAL MEASURES WHICH THE LATTER COMPRISE HAVE A VALID TECHNICAL BASIS. THIS APPROACH, IT MUST BE ADMITTED, HAS NOT BEEN ALWAYS ADOPTED IN THE SECTORS OF INDUSTRY PRODUCING POLLUTANTS. THE USE OF NUCLEAR ENERGY ALSO PRODUCES MATERIALS, SOME OF WHICH CAN BE RECOVERED AND BURNED AS FUEL, SOURCES OF RADIATION FOR THERAPY, ETC. OTHERS, AT LEAST ON THE BASIS OF THE PRESENT TECHNOLOGY, ARE UNUSABLE: THESE PRODUCTS CONSTITUTE RADIOACTIVE WASTE. THIS WASTE HAS TWO ESSENTIAL CHARACTERISTICS: (1) RADIOACTIVITY AND (2) CARCINOGENICITY.

ORDS   RISK;WASTE DISPOSAL;SURVEY;COST ANALYSIS;REGULATION;WASTE
              MANAGEMENT

/07/0000001-00001157/   31
ESSION NO.   00X0166860
LE   SUMMARY OF THE RISK ASSESSMENT MADE OF THE TRANSPORT OF
              PLUTONIUM NITRATE
THOR(S)   CHICKEN JC
RPAUTH   UKAEA SAFETY & RELIABILITY DIRECTORATE, U.K.
TE   1980
PE   R
MO   SRD R 197 + UKRSR-356 +. 8 PPS, 2 FIGS, 2 REFS, OCT. 1980
AIL   AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH,
              DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S.
              NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
TEGORY   030000;250000
ITION   0130
RP CODE   UKA
UNTRY   U
STRACT   PAPER PRESENTS AN OVERALL VIEW OF THE RISKS INVOLVED IN
              TRANSPORTING PLUTONIUM NITRATE FROM DOUNREAY TO WINDSCALE. THE
              PRESENTATION IS DIVIDED INTO SEVEN DISTINCT PARTS WHICH ARE
              MOVEMENT, CRITERIA, PROBABILITY OF RUPTURE, CONSEQUENCES OF
              RELEASE, MONITORING, DECONTAMINATION AND CONCLUSIONS.
YWORDS   UNITED KINGDOM;PLUTONIUM;BENEFIT VS RISK;FOREIGN EXCHANGE;
              DOUNREAY (LMFBR);WINDSCALE;DECONTAMINATION;RISK

//07/0000001-00001157/   32
ESSION NO.   00D0166815
LE   RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE:
              SENSITIVITY ANALYSIS OF THE ENVIRONMENTAL TRANSPORT MODEL
THOR(S)   HELTON JC;IMAN RL
RPAUTH   SANDIA NATIONAL LABS., ALBUQUERQUE, NM
TE   1980
PE   A
MO   NUREG/CR-1636(VOL. 2) + SAND 79-1393 +. 225 PPS, TABS, FIGS,
              REFS, DEC. 1980
AIL   AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S.
              DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY   230000;140000
TION   0130
P CODE   AUA
NTRY   A
STRACT   RESULTS ARE PRESENTED FROM A SENSITIVITY ANALYSIS STUDY MODEL
              DEVELOPED TO REPRESENT THE SURFACE MOVEMENT OF RADIONUCLIDES.
              THE STUDY HAS TWO PURPOSES: (1) TO DEVELOP A CAPABILITY TO
              PERFORM SENSITIVITY ANALYSIS OF THE MODEL AND (2) INQUIRE
              INSIGHT WITH RESPECT TO VARIABLES WHICH INFLUENCE PREDICTIONS
              OF THE MODEL. THE FOLLOWING INDEPENDANT VARIABLES ARE
              INTRODUCED: RIVER SYSTEMS, AND RADIONUCLIDE PROPERTIES.
              DEPENDENT VARIABLES ARE: (1) RADIONUCLIDE CONCENTRATION, (2)
              DISSOLVED RADIONUCLIDES AND (3) RADIONUCLIDE CONCENTRATION IN
              SURFACE WATER.
WORDS   RISK;NUMERICAL METHOD;WASTE DISPOSAL;ENVIRONMENT;ANALYTICAL
              MODEL;GEOLOGY;HJCK;NRC-AN

/07/0000001-00001157/   33
ESSION NO.   00J0166811
LE   EPRI NUCLEAR FUEL-CYCLE ACCIDENT RISK ASSESSMENT
E   1981
E   U
O   7 PPS, 5 FIGS, 11 REFS, NUCLEAR SAFETY, 22(3), PP. 300-306
              (MAY-JUNE 1981)
EGORY   230000;130000
TION   0130
NTRY   A
B   NUSA
TRACT   THE PRESENT RESULTS OF THE NUCLEAR FUEL-CYCLE ACCIDENT RISK
              ASSESSMENT CONDUCTED BY THE ELECTRIC POWER RESEARCH INSTITUTE
              SHOW THAT THE TOTAL RISK CONTRIBUTION OF THE NUCLEAR FUEL CYCLE
              IS ONLY APPROX. 1% OF THE ACCIDENT RISK OF THE POWER PLANT;
              HENCE, WITH LITTLE ERROR, THE ACCIDENT RISK OF NUCLEAR ELECTRIC
              POWER IS ESSENTIALLY THAT OF THE POWER PLANT ITSELF. THE
              POWER-PLANT RISK, ASSUMING A VERY LARGE USAGE OF NUCLEAR POWER

BY THE YEAR 2005, IS ONLY APPROX. 0.5% OF THE RADIOLOGICAL RISK OF NATURAL BACKGROUND. THE SMALLNESS OF THE FUEL-CYCLE RISK RELATIVE TO THE POWER-PLANT RISK MAY BE ATTRIBUTED TO THE LACK OF INTERNAL ENERGY TO DRIVE AN ACCIDENT AND THE SMALL AMOUNT OF DISPERSIBLE MATERIAL. THIS WORK AIMS AT A REALISTIC ASSESSMENT OF THE PROCESS HAZARDS, THE EFFECTIVENESS OF CONFINEMENT AND MITIGATIONS SYSTEMS AND PROCEDURES, AND THE ASSOCIATED LIKELIHOOD OF ERRORS AND THE ESTIMATED SIZE OF ERRORS.

YWORDS RISK;FUEL CYCLE;ACCIDENT;EPRI;MINING;MILLING;FUEL REPROCESSING; ELECTRIC POWER;TRANSPORTATION AND HANDLING

7/07/0000001-00001157/ 34

CESSION NO. 00J0166809
TLE NUCLEAR POWER: PUTTING THE RISKS INTO PERSPECTIVE
THOR(S) WHEATLEY BM;MAYNEORD WV
RPAUTH UNIV. OF LONDON, U.K.
TE 1981
PE D
RD 10 PPS, 9 FIGS, CEGB RESEARCH, NO. 11, PP. 31-40 (JAN. 1981)
TEGORY 230000
ITION 0130
NTRY U
STRACT DAMAGE TO HEALTH COULD IN PRINCIPLE BE CAUSED BY RADIATION, PERHAPS FROM RADIOACTIVE MATERIALS IN MINUTE QUANTITIES. IT IS THEREFORE IMPORTANT IN RELATION TO POWER REACTORS TO SHOW THAT LEVELS OF RADIATION AND RADIOACTIVITY ARE DEMONSTRABLY BELOW ACCEPTABLE LEVELS. THIS CONSIDERATION REQUIRES A DEFINITION OF WHAT IS 'ACCEPTABLE' SO THAT PLANT CAN BE CORRESPONDINGLY DESIGNED, AND LATER INVOLVES THE ESTABLISHMENT OF AN ADEQUATE MONITORING PROGRAM. SENSITIVE INSTRUMENTS CHECK THAT STATION WORKERS RECEIVE ONLY LOW DOSES AND THAT THE PUBLIC ENVIRONMENT IS ALSO PROTECTED. THE FULFILLMENT OF THESE RESPONSIBILITIES REQUIRES RESEARCH IN MATHEMATICAL, PHYSICAL AND BIOLOGICAL SCIENCES. THIS ARTICLE OUTLINES ONE OR TWO EXAMPLES OF RELEVANT RESEARCH IN THIS FIELD AT BERKELEY NUCLEAR LABORATORIES. (FAH)
YWORDS UNITED KINGDOM;N-POWER, SAFETY OF;RISK;R AND D PROGRAM; RADIATION IN PERSPECTIVE;DOSE;CANCER

/07/0000001-00001157/ 35

CESSION NO. 00X0166808
LE ON THE RELATION OF VARIOUS RELIABILITY MEASURES TO EACH OTHER AND TO GAME THEORETIC VALUES
HOR(S) STRIP DR
PAUTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM
E 1981
E N
D NUREG/CR-1800 + SAND 80-2624 +. 33 PPS, 2 FIGS, 10 REFS, JAN. 1981
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 220000
TION 0130
P CODE ADA
NTRY A
TRACT A VARIETY OF MEASURES HAVE RECENTLY BEEN PROPOSED FOR MEASURING THE RELATIVE IMPORTANCE OF INDIVIDUAL COMPONENTS IN THE OVERALL RELIABILITY OF A SYSTEM. SEVERAL OF THESE SEEMINGLY DIFFERENT MEASURES ARE VERY CLOSELY RELATED UNDER THE CONDITIONS TYPICALLY ASSUMED IN THE RELIABILITY LITERATURE. THE MEASURES ARE ALSO CLOSELY RELATED TO THE PROBABILISTIC VALUES OF GAME THEORY.
WORDS RELIABILITY, COMPONENT;RISK;ANALYTICAL TECHNIQUE;GAME THEORY; HJCK;NRC-AN;RELIABILITY ANALYSIS

/07/0000001-00001157/ 36

SSION NO. 00X0166807
LE RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE: ASYMPTOTIC PROPERTIES OF THE ENVIRONMENTAL TRANSPORT MODEL
HOR(S) HELTON JC;IMAN RL;BROWN JB
AUTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM
1981
N
NUREG/CR-1636(VOL. 3) + SAND 79-1908 +. 173 PPS, TABS, FIGS,

FEB. 1981
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 230000;140000
TION 0130
RP CODE AOA
UNTRY A
STRACT THE ENVIRONMENTAL TRANSPORT MODEL IS A COMPARTMENTAL MODEL DEVELOPED TO REPRESENT THE SURFACE MOVEMENT OF RADIONUCLIDES. THE PURPOSE OF THE PRESENT STUDY IS TO INVESTIGATE THE ASYMPTOTIC BEHAVIOR OF THE MODEL AND TO ACQUIRE INSIGHT WITH RESPECT TO SUCH BEHAVIOR AND THE VARIABLES WHICH INFLUENCE IT. FOR FOUR VARIATIONS OF A HYPOTHETICAL RIVER RECEIVING A RADIONUCLIDE DISCHARGE, THE FOLLOWING PROPERTIES ARE CONSIDERED: PREDICTED ASYMPTOTIC VALUES FOR ENVIRONMENTAL RADIONUCLIDE CONCENTRATIONS AND TIME REQUIRED FOR ENVIRONMENTAL RADIONUCLIDE CONCENTRATIONS TO REACH 90% OF THEIR PREDICTED ASYMPTOTIC VALUES. INDEPENDENT VARIABLES OF TWO TYPES ARE USED TO DEFINE EACH VARIATION OF THE RIVER: VARIABLES WHICH DEFINE PHYSICAL PROPERTIES OF THE RIVER SYSTEM (E.G., SOIL DEPTH, RIVER DISCHARGE AND SEDIMENT RESUSPENSION) AND VARIABLES WHICH SUMMARIZE RADIONUCLIDE PROPERTIES (I.E., DISTRIBUTION COEFFICIENTS). SENSITIVITY ANALYSIS TECHNIQUES BASED ON STEPWISE REGRESSION ARE USED TO DETERMINE THE DOMINANT VARIABLES INFLUENCING THE BEHAVIOR OF THE MODEL.
YWORDS RISK;NUMERICAL METHOD;GEOLOGY;WASTE DISPOSAL;ANALYTICAL MODEL; ENVIRONMENT;TRANSPORT;HJCK;SOIL, RADIONUCLIDE MOVEMENT THROUGH; NRC-AN

7/07/0000001-0000011577 37
CESSION NO. 00X0166605
TLE RISK METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTE: MODEL DESCRIPTION AND USER MANUAL FOR PATHWAYS MODEL
THOR(S) HELTON JC;KAESTNER PC
RPAUTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM
TE 1981
PE R
OU NUREG/CR-1636(VOL. 1) + SAND 78-1711 +. 170 PPS, TABS, FIGS, MARCH 1981
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 230000;140000
TION 0130
RP CODE AOA
UNTRY A
STRACT A MODEL FOR THE ENVIRONMENTAL MOVEMENT AND HUMAN UPTAKE OF RADIONUCLIDES IS PRESENTED. THIS MODEL IS DESIGNATED THE PATHWAYS-TO-MAN MODEL AND WAS DEVELOPED AS PART OF A PROJECT FUNDED BY THE NUCLEAR REGULATORY COMMISSION TO DESIGN A METHODOLOGY TO ASSESS THE RISK ASSOCIATED WITH THE GEOLOGIC DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTE. THE PATHWAYS-TO-MAN MODEL IS DIVIDED INTO TWO SUBMODELS. ONE OF THESE, THE ENVIRONMENTAL TRANSPORT MODEL, REPRESENTS THE LONG-TERM DISTRIBUTION AND ACCUMULATION OF RADIONUCLIDES IN THE ENVIRONMENT. THIS MODEL IS BASED ON A MIXED-CELL APPROACH AND DESCRIBES RADIONUCLIDE MOVEMENT WITH A SYSTEM OF LINEAR DIFFERENTIAL EQUATIONS. THE OTHER, THE TRANSPORT-TO-MAN MODEL, REPRESENTS THE MOVEMENT OF RADIONUCLIDES FROM THE ENVIRONMENT TO MAN. THIS MODEL IS BASED ON CONCENTRATION RATIOS. GENERAL DESCRIPTIONS OF THESE MODELS ARE PROVIDED IN THIS REPORT. FURTHER, DOCUMENTATION IS PROVIDED FOR THE COMPUTER PROGRAM WHICH IMPLEMENTS THE PATHWAYS MODEL.
WORDS RISK;NUMERICAL METHOD;GEOLOGY;WASTE DISPOSAL;ANALYTICAL MODEL; PROCEDURES AND MANUALS;ENVIRONMENT;RADIONUCLIDE UPTAKE;HJCK; NRC-AN;SOIL, RADIONUCLIDE MOVEMENT THROUGH;SOIL, RADIONUCLIDE MOVEMENT THROUGH

7/07/0000001-0000011577 38
ESSION NO. 00X0166798
LE SENSITIVITY OF RISK PARAMETERS TO HUMAN ERRORS IN REACTOR SAFETY STUDY FOR A PWR
HOR(S) SAMANTA PK;HALL RE;SWOBODA AL
PAUTH BROOKHAVEN NATIONAL LAB., UPTON, NY
E 1981

N

D NUREG/CR-1879 + ORNL-NUREG-51322 +. 125 PPS. FIGS. REFS. JAN. 1981

IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

EGORY 220000;010000

ITION 0130

AP CODE D2A

UNTRY A

STRACT SENSITIVITIES OF THE RISK PARAMETERS, EMERGENCY SAFETY SYSTEM UNAVAILABILITIES, ACCIDENT SEQUENCE PROBABILITIES, RELEASE CATEGORY PROBABILITIES AND CORE MELT PROBABILITY WERE INVESTIGATED FOR CHANGES IN THE HUMAN ERROR RATES WITHIN THE GENERAL METHODOLOGICAL FRAMEWORK OF THE REACTOR SAFETY STUDY (RSS) FOR A PRESSURIZED WATER REACTOR (PWR). IMPACT OF INDIVIDUAL HUMAN ERRORS WERE ASSESSED BOTH IN TERMS OF THEIR STRUCTURAL IMPORTANCE TO CORE MELT AND RELIABILITY IMPORTANCE ON CORE MELT PROBABILITY. THE HUMAN ERROR SENSITIVITY ASSESSMENT OF A PWR (HESAP) COMPUTER CODE WAS WRITTEN FOR THE PURPOSE OF THIS STUDY. CORE MELT PROBABILITY PER REACTOR YEAR SHOWED SIGNIFICANT INCREASE WITH THE INCREASE IN THE HUMAN ERROR RATES, BUT DID NOT SHOW SIMILAR DECREASE WITH THE DECREASE IN THE HUMAN ERROR RATES DUE TO THE DOMINANCE OF THE HARDWARE FAILURES.

YWORDS RISK;REACTOR, PWR;SYSTEM ANALYSIS;COMPUTER PROGRAM;RELIABILITY ANALYSIS;FAULT TREE ANALYSIS;CORE MELTDOWN;FAILURE, OPERATOR ERROR;ACCIDENT, PROBABILITY OF;SAFETY ANALYSIS;HJCK;NRC-RG; NRC-XA

770/0000001-00001577 39

CESSION NO. 00X0166796

LE REDUCTION IN REACTOR RISK BY THE MITIGATION OF ACCIDENT CONSEQUENCES

THOR(S) DENNING RS;CYBULSKIS P

RPAUTH BATTELLE COLUMBUS LABS., OHIO

TE 1981

PE N

PU 8 PPS, 3 TABS, 1 FIG, 24 REFS. NUCLEAR SAFETY, 22(2), PP. 165-72 (MARCH-APRIL 1981)

EGORY 220000

TION 0130

P CODE CAF

TRACT APPROACHES ARE EXAMINED TO REDUCING THE PUBLIC RISK FROM NUCLEAR POWER REACTORS BY USING SAFETY SYSTEMS TO CONTROL THE CONSEQUENCES OF CORE MELTDOWN ACCIDENTS. MECHANISMS THAT COULD LEAD TO FAILURE OF THE CONTAINMENT ARE IDENTIFIED. ENGINEERED SAFETY FEATURES ARE THEN DESCRIBED WHICH WOULD REDUCE THE LIKELIHOOD OF CONTAINMENT FAILURE OR REDUCE THE SUBSEQUENT RELEASE OF RADIOACTIVITY TO THE ENVIRONMENT. THE POTENTIAL FOR REDUCING RISK IS ASSESSED AND IS FOUND TO BE SIGNIFICANT FOR SOME SAFETY FEATURES. HOWEVER, TO ADEQUATELY ASSESS THE POTENTIAL REDUCTION IN RISK, A SYSTEMATIC ANALYSIS OF SYSTEM INTERACTIONS MUST BE PERFORMED.

WORDS ACCIDENT, CORE DISRUPTIVE;ACCIDENT, CONSEQUENCES;RISK;HAZARDS ANALYSIS;CORE MELTDOWN;SAFETY ANALYSIS;ENVIRONMENT

70/0000001-00001577 40

ESSION NO. 00X0166795

LE RISK, FEAR AND PUBLIC SAFETY

HOR(S) SIDDALL E

RPAUTH ATOMIC ENERGY OF CANADA LTD.,

E 1981

E N

D 50 PPS, APRIL 1981

IL AVAILABILITY - ATOMIC ENERGY OF CANADA LTD., ENGINEERING CO., SHERIDAN PARK RESEARCH COMMUNITY, MISSISSAUGA, ONTARIO L5K1B2

EGORY 220000

TION 0130

NTRY C

TRACT A RATIONAL APPROACH TO PUBLIC SAFETY, INTRODUCED IN PREVIOUS WORK, IS DEVELOPED. IT BRINGS OUT THE FACTORS WHICH WOULD PERMIT A REASONED ASSESSMENT OF THE TRUE OVERALL IMPACT OF TECHNOLOGICAL ACTIVITIES ON SAFETY IN A PRESENT DAY COMMUNITY. A METHOD OF OPTIMIZING THE OVERALL SAFETY POLICY AND PROGRAM IN

A SOCIETY, PRIMARILY FROM THE VIEWPOINT OF THE CITIZEN RATHER THAN THE SOURCE OF RISK, IS OUTLINED. THE TRUE IMPACT OF A TYPICAL TECHNOLOGICAL ACTIVITY ON PUBLIC SAFETY IS SHOWN AS A MODEL. FROM THIS IT CAN BE SEEN THAT THERE IS AN INESCAPABLE NEED FOR AN UNBIASED QUANTITATIVE ASSESSMENT OF BOTH RISKS AND SAFETY BENEFITS, UPON WHICH SAFETY DECISIONS CAN BE BASED. PARTICULAR QUANTITATIVE ASPECTS OF RISK AND SAFETY BENEFITS IN OUR SOCIETIES ARE DEVELOPED IN SIX APPENDICES.

WORDS COST BENEFIT;HAZARDS ANALYSIS;RISK;SAFETY REVIEW; SOCIO/PHILOSOPHICAL CONSIDERATION

/0/0000001-00001157/ 41

ESSION NO. 00X0166794

LE RISK IN A COMPLEX SOCIETY: A MARSH & MCLENNAN PUBLIC OPINION SURVEY

AUTH M&M NUCLEAR CONSULTANTS, NY

1980

N

66 PPS, BULLETIN NO. 5, JUNE 1980

IL AVAILABILITY - M & M NUCLEAR CONSULTANTS, 1221 AVENUE OF THE AMERICAS, NEW YORK, NY 10020

EGORY 250000;010000

TION 0130

NTRY U

TRACT A STUDY DONE BY LOUIS HARRIS (ET. AL) USING PUBLIC CORPORATION OFFICES, INVESTORS, MEMBERS OF CONGRESS, THE FEDERAL AGENCIES AND THE PUBLIC TO DETERMINE THEIR ATTITUDE ON RISK, CONTROL OF RISK, ENERGY AND RISK, SAFETY, AND COMPARISON OF RISK AND INVESTMENT TABLES.

WORDS RISK;SURVEY;SOCIO/PHILOSOPHICAL CONSIDERATION;DATA COLLECTION; COMPARISON;MORTALITY

/0/0000001-00001157/ 42

ESSION NO. 00X0166793

LE COMPARATIVE RISKS OF ELECTRICITY PRODUCTION SYSTEMS: A CRITICAL SURVEY OF THE LITERATURE

AUTH HEALTH & SAFETY EXECUTIVE, U.K.

1980

N

31 PPS, RESEARCH PAPER NO. 11, DEC. 1980

IL AVAILABILITY - HER MAJESTY'S STATIONERY OFFICE, 49 HIGH HOLBORN, LONDON WC1V 6HB, ENGLAND

EGORY 230000

TION 0130

NTRY U

TRACT EXPERIENCE TO DATE HAS LED TO THE CONCLUSION THAT ANY ENERGY SOURCE CURRENTLY BEING SERIOUSLY CONSIDERED FOR THE LARGE-SCALE GENERATION OF ELECTRICITY CAN BE MADE SAFE ENOUGH TO ALLOW ITS USE. THE SURVEY CONCENTRATES ON THE THREE 'CONVENTIONAL' ELECTRICITY SYSTEMS: COAL, OIL, AND NUCLEAR, AND EXAMINES THE RISKS OF WHOLE PRODUCTION SYSTEMS, FROM THE CONSTRUCTION OF PLANT AND THE MINING OF FUEL TO THE DISPOSAL OF WASTE. THE CRITICAL ANALYSIS SHOWS THAT IT IS MISLEADING TO MAKE A SIMPLE FORM OF COMPARISON OF RISK: MORE COMPLEX COMPARISONS ARE NECESSARY. THE SURVEY COMPARES PUBLISHED ASSESSMENTS OF THE RISKS AT EACH STAGE OF THE ELECTRICITY PRODUCTION PROCESS, TABULATING THE RISKS PER UNIT OF ELECTRICITY PRODUCED WHERE QUANTITATIVE ESTIMATES EXIST IN THE LITERATURE, WITH APPROPRIAATE CAUTIONARY REMARKS WHERE THE DATA ARE QUESTIONABLE. DISP 166793 ABST

WORDS RISK;SURVEY;DATA COLLECTION;POWER PLANT, NUCLEAR;POWER PLANT, FOSSIL FUEL;COMPARISON

0/0000001-00001157/ 43

SSION NO. 00X0166792

E ENERGY ALTERNATIVES-A COMPARATIVE ANALYSIS

AUTH UNIV. OF OKLAHOMA, NORMAN

1975

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PB 246365 + FEA/D-75/661 +. APPROX. 500 PPS, MAY 1975

L AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S. DEPT. OF COMMERCE, SPRINGFIELD, VA. 22161

GORY 230000;010000

ION 0130

CODE UCD
NTRY A
TRACT THIS REPORT CONTRIBUTES TO THE DEVELOPMENT OF A METHODOLOGY FOR SYSTEMATICALLY IDENTIFYING, ASSESSING, AND COMPARING ENERGY ALTERNATIVES. IT PROVIDES DESCRIPTIONS AND DATA ON THE MAJOR ENERGY RESOURCE SYSTEMS IN THE UNITED STATES AND SUGGESTS PROCEDURES FOR USING THIS DESCRIPTIONS AND DATA. THE STUDY CONSISTS OF TWO PARTS; THE FIRST CONTAINS DESCRIPTIONS OF COAL, AND SHALE, CRUDE OIL, NATURAL GAS, TAR SANDS, NUCLEAR FISSION, NUCLEAR FUSION, GEOTHERMAL, HYDROELECTRICITY, ORGANIC WASTES AND SOLAR ENERGY SYSTEMS. PART TWO DESCRIBES PROCEDURES FOR USING THE DESCRIPTIONS AND DATA IN SYSTEMATICALLY EVALUATING AND COMPARING ENERGY ALTERNATIVES.
YWORDS RISK;COMPARISON;SAFETY EVALUATION;DATA COLLECTION;ENERGY POLICY; POWER PLANT, NUCLEAR;POWER PLANT, FOSSIL FUEL;FUSION;POWER PLANT, HYDROELECTRIC;HAZARDS ANALYSIS

7/07/0000001-00001577 44
CESSION NO. 00X0166791
TLE NUCLEAR AND NON-NUCLEAR RISK-AN EXERCISE IN COMPARABILITY
RPAUTH POLLUTION PREVENTION CONSULTANTS LTD., CRAWLEY, ENGLAND
TE 1980
PE N
MO EUR-6417EN +. 355 PPS, 1980
AIL AVAILABILITY - EUROPEAN COMMUNITY INFORMATION SERVICE, 2100 M ST., N.W., SUITE 707, WASHINGTON, D.C. 20027
TEGORY 230000
ITION 0130
UNTRY L
STRACT WITH THE INCREASING EMPHASIS WHICH IS BEING PLACED ON THE USE OF RELIABILITY ENGINEERING TECHNIQUES AS A MEANS OF IMPROVING THE DESIGN OF NUCLEAR POWER PLANTS, PARTICULARLY WITH RESPECT TO SAFETY, THE DEFINITION OF QUANTITATIVE DESIGN TARGETS IN TERMS OF 'ACCEPTABLE RISK' HAS BECOME A MATTER OF IMPORTANCE. A POSSIBLE METHOD OF ESTABLISHING SUCH TARGETS IS TO EXAMINE THE NON-NUCLEAR RISKS WHICH SOCIETIES ACCEPT AND TO EVALUATE THEIR COMPARABILITY WITH THE NUCLEAR RISKS. CONSEQUENTLY THE MAIN OBJECTIVES OF THIS STUDY ARE TO EXAMINE THE UNDERLYING FACTORS WHICH DETERMINE THE EXISTING LEVELS OF NON-NUCLEAR RISKS AND TO SUGGEST MEANS OF WORKING TOWARDS THE DEVIATION. OF 'ACCEPTABLE NUCLEAR RISKS' WHICH WILL BE COMPATIBLE WITH NON-NUCLEAR RISKS. THE CAUSE STUDIES ARE BASED IN U.K. EXPERIENCE.
YWORDS RISK;COMPARISON;HAZARDS ANALYSIS;SAFETY ANALYSIS;SAFETY PRINCIPLES AND PHILOSOPHY;SAFETY REVIEW;ANALYTICAL MODEL; SENSITIVITY ANALYSIS;SOCIO/PHILOSOPHICAL CONSIDERATION

7/07/0000001-00001577 45
CESSION NO. 00X0166790
TLE EXECUTIVE SEMINAR ON THE FUTURE ROLE OF RISK ASSESSMENT AND RELIABILITY ENGINEERING IN NUCLEAR REGULATION
THOR(S) VAN ERP JB
PAUTH U.S. NUCLEAR REGULATORY COMMISSION ; ARGONNE NATIONAL LAB., IL
E 1981
E N
O NUREG/CP-0017 + ANL-81-3 +. 145 PPS, MARCH 1981
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 230000
TION 0130
P CODE NRC;CZA
NTRY A
TRACT THE EXECUTIVE SEMINAR WAS HELD IN THE AFTERMATH OF THE THREE MILE ISLAND ACCIDENT TO HELP THE NUCLEAR REGULATORY COMMISSION STAFF TO UNDERSTAND THE STRENGTHS AND WEAKNESSES OF PROBABILISTIC RISK ASSESSMENTS AND TO UNDERSTAND THE CURRENT ATTITUDES ON ITS USE IN NUCLEAR SAFETY REGULATION. SIX PAPERS AND PANEL DISCUSSIONS ARE INCLUDED.
WORDS RISK;HAZARDS ANALYSIS;SAFETY ANALYSIS;SOCIO/PHILOSOPHICAL CONSIDERATION;SENSITIVITY ANALYSIS;REGULATION;COMPARISON;HJCK; NRC-1

/0/0000001-00001577 46
SSION NO. 00X0166789
LE OCCUPATIONAL SAFETY DATA AND CASUALTY RATES FOR THE URANIUM

FUEL CYCLE
HOR(S)   HOY HC;O'DONNELL FR
PAUTH   OAK RIDGE NATIONAL LAB., TN
E   1981
E   N
    ORNL-5797 +. VP, AUG. 1981
IL   AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY   230000;130000
TION   0130
P CODE   F2C
NTRY   A
RACT   OCCUPATIONAL CASUALTY (INJURIES, ILLNESSES, FATALITIES, AND LOST WORKDAYS) AND PRODUCTION DATA ARE PRESENTED AND ARE USED TO CALCULATE OCCUPATIONAL CASUALTY INCIDENCE RATES FOR TECHNOLOGIES THAT THE URANIUM FUEL CYCLE, INCLUDING: MINING, MILLING, CONVERSION, AND ENRICHMENT OF URANIUM; FABRICATION OF REACTOR FUEL; TRANSPORTATION OF URANIUM AND FUEL ELEMENTS; GENERATION OF ELECTRIC POWER; AND TRANSMISSION OF ELECTRIC POWER. EACH TECHNOLOGY IS TREATED IN A SEPARATE CHAPTER. ALL DATA SOURCES ARE REFERENCED. ALL STEPS USED TO CALCULATE NORMALIZED OCCUPATIONAL CASUALTY INCIDENCE RATES FROM THE DATA ARE PRESENTED. RATES GIVEN INCLUDE FATALITIES, SERIOUS CASES, AND LOST WORKDAYS PER 100 MAN-YEARS WORKED, PER 10(12) BTU OF ENERGY OUTPUT, AND PER OTHER APPROPRIATE UNITS OF OUTPUT.
WORDS   FUEL CYCLE;RISK;HAZARDS ANALYSIS;URANIUM,DATA COLLECTION

/07/0000001-0000115//   47
ESSION NO.   00X0166787
LE   PLAN FOR DEVELOPING A SAFETY GOAL
PAUTH   U.S. NUCLEAR REGULATORY COMMISSION
E   1980
E   N
    NUREG-0735 +. 20 PPS, OCT. 1980
IL   AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY   230000;170000;180000;010000
TION   0130
P CODE   NRC
NTRY   A
TRACT   THIS REPORT PRESENTS A COMMISSION APPROVED PLAN FOR DEVELOPMENT AND/OR ARTICULATION OF NRC SAFETY OBJECTIVES, NOTABLY, BUT NOT EXCLUSIVELY, WITH RESPECT TO POWER REACTORS. ATTENTION IS GIVEN TO INTERIM OBJECTIVES OBTAINABLE IN THE NEAR TERM, BUT ALSO INCLUDING OVER ALL PROGRAM CONSIDERATIONS AS WELL. THE POLICY ARTICULATION TO BE DEVELOPED WILL INCLUDE SOME GENERAL APPROACH TO RISK ACCEPTABILITY AND SAFETY-COST TRADE OFFS, QUANTATIVE SAFETY GOALS, SAFETY IMPROVEMENT GOALS AND STANDARDS FOR REVIEW OF PAST ACTIONS IN VIEW OF NEW RULES AND IMPROVED PRACTICES.
WORDS   RISK;SAFETY ANALYSIS;SAFETY EVALUATION;SAFETY PROGRAM;SAFETY REVIEW;ACCIDENT ANALYSIS;COMPARISON, THEORY AND EXPERIENCE; HAZARDS ANALYSIS;AGENCY, NRC;SAFETY PRINCIPLES AND PHILOSOPHY

/07/0000001-0000115//   48
ESSION NO.   00X0166788
LE   "CANVEY" AN INVESTIGATION OF POTENTIAL HAZARDS FROM OPERATIONS IN THE CANVEY ISLAND/THURROCK AREA
PAUTH   UK HEALTH & SAFETY COMMISSION, LONDON
E   1978
E   N
    192 PPS, 1978 (ISBN 011 883200 X)
IL   AVAILABILITY - HER MAJESTY'S STATIONERY OFFICE, 49 HIGH HOLBORN, LONDON WC1V 6HB, ENGLAND
EGORY   230000
TION   0130
NTRY   U
TRACT   AT THE REQUEST OF THE SECRETARIES OF STATE AND OF THE ENVIRONMENT THE RISKS TO HEALTH AND SAFETY ASSOCIATED WITH VARIOUS INSTALLATIONS, BOTH EXISTING AND PROPOSED, ON CANVEY ISLAND AND THE NEIGHBORING PART OF THURROCK. THIS AREA CONTAINS SEVERAL PETROCHEMICAL PROCESSING AND STORAGE FACILITIES AS WELL AS TANK FARMS FOR LIQUEFIED GAS AND IS USED AS A RECEIVING AND STAGING POINT FOR STEAMSHIP, RAILWAY AND

MOTOR TRANSPORT. IT WAS CONCLUDED THAT THE RISK OF ACCIDENTAL FATALITIES ARE ABOUT 5 IN 10,000 PER YEAR WHICH IS ABOUT THE SAME AS THE REST OF THE POPULATION IN THE 23-34 GROUP AND THAT IF THE RECOMMENDED IMPROVEMENTS ARE MADE THAT THE INSTALLATION OF ANOTHER FACILITY WILL NOT GREATLY INCREASE THE HAZARD.

RDS RISK;HAZARDS ANALYSIS;COMPARISON;REPORT, SAR;SAFETY EVALUATION; SITING, CHEMICAL PROCESS PLANT

)/0000001-0000115// 49
SSION NO. 00X0166765
E A RISK COMPARISON
R(S) HALL RE;COPPOLA A
UTH BROOKHAVEN NATIONAL LAB., UPTON, NY
1981
N
NUREG/CR-1916 + BNL-NUREG-51338 +. 90 PPS. 16 TABS. 21 FIGS. JAN. 1981
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GORY 230000;010000
ION 0130
CODE S2A
RY A
ACT PRESENTS DATA FOR THE COMPARISON OF SOCIETAL RISK FROM NATURAL AND MAN-MADE HAZARDS. ONLY FATALITIES RESULTING FROM THE HAZARDS ARE USED IN THE COMPARISON, WITH THE DATA AND THE COMPARATIVE ANALYSIS TAKEN FROM CURRENT LITERATURE. IN COMPARING SOCIETAL RISKS FOR MOST OF THE HAZARDS, BOTH EXPECTED VALUES AND FREQUENCY VS. CONSEQUENCE CURVES ARE PRESENTED. FOR A SUBSET OF HAZARDS, NOTABLY THE POWER GENERATION TECHNOLOGIES (NUCLEAR, COAL, OIL, AND GAS), WHICH HAVE NOT EXHIBITED HIGH CONSEQUENCE EVENTS (CATASTROPHES), THE COMPARISONS ARE BASED ON ESTIMATED EXPECTED VALUES ONLY. INDIVIDUAL RISK DATA ARE PRESENTED IN TWO WAYS, A PROBABILITY OF DEATH WITHIN A YEAR AND THE AMOUNT OF LIFE SHORTENING OF AN AVERAGE LIFE SPAN.
RDS RISK;MORTALITY;COMPARISON;DATA COLLECTION;SAFETY EVALUATION; SOCIO/PHILOSOPHICAL CONSIDERATION;HJCK;NRC-7;NRC-RG;HAZARDS ANALYSIS;MORTALITY

)/0000001-0000115// 50
SSION NO. 0000166764
E WASH-1400: QUANTIFYING THE UNCERTAINTIES
UTH SCIENCE APPLICATIONS INC., PALO ALTO, CA ; ELECTRIC POWER RESEARCH INST., PALO ALTO, CA
1981
U
7 PPS. 4 TABS, 1 FIG. 8 REFS. NUCLEAR TECHNOLOGY, 53(3), PP. 374-80 (JUNE 1981)
GORY 230000
ION 0130
CODE SAI;EPR
RY A
ACT FOR THE SPECIFIC REACTORS EVALUATED IN THE REACTOR SAFETY STUDY (WASH-1400), QUANTIFIED UNCERTAINTIES IN THE CALCULATED CONSEQUENCES AND PROBABILITIES WERE REPORTED, ALONG WITH MEDIAN ESTIMATES. THE WASH-1400 EVALUATION OF EARLY FATALITIES GAVE THESE UNCERTAINTIES AS MULTIPLICATIVE FACTORS OF (1/4, 4) ON CONSEQUENCES AND (1/5, 5) ON PROBABILITIES. ACCOUNTING FOR FACTORS THAT WERE NOT CONSIDERED, THESE UNCERTAINTIES ARE BETTER STATED AS MULTIPLICATIVE FACTORS OF (1/15, 15) AND 1/20, 20) FOR CONSEQUENCES AND PROBABILITIES, RESPECTIVELY. IN ADDITION TO THIS CHANGE IN UNCERTAINTY, THE MEDIAN VALUES OF EARLY FATALITIES REPORTED IN WASH-1400 MAY BE TOO HIGH BY FACTORS OF 5 FOR CONSEQUENCES AND 12 FOR PROBABILITIES. THUS, A NEW UPPER BOUND IS FOUND THAT IS LESS THAN THAT STATED IN WASH-1400.
RDS RISK;HAZARDS ANALYSIS;SAFETY ANALYSIS;SAFETY REVIEW;COMPARISON

)/0000001-0000115// 51
SSION NO. 00X0166783
E ISSUES AND PROBLEMS IN INFERRING A LEVEL OF ACCEPTABLE RISK
R(S) SALEM SL;SOLOMON KA;YESLEY MS
UTH THE RAND CORP., SANTA MONICA, CA
1980

+
R-2561-DOE +. 110 PPS. 13 TABS. 5 FIGS. AUG. 1980
AVAILABILITY - PUBLICATIONS DEPT.. THE RAND CORP.. 1700 MAIN
ST.. SANTA MONICA. CALIF. 90406
RY 230000
N 0130
Y A
CT AS A MODERN SOCIETY BECOMES INCREASINGLY DEPENDENT ON BENEFITS OF TECHNOLOGICAL EXPERTISE. IT FACES PROBLEMS OF REDUCING THE HARMS AND RISK THAT INEVITABLY ACCOMPANY THESE TECHNOLOGICAL INNOVATIONS. THIS STUDY ADDRESS'S TWO PARTS OF THAT TRADE-OFF: FORMULATION OF RISK ACCEPTANCE CRITERIA AND MANAGEMENT OF RISK BY GOALS. ELEVEN ENERGY ALTERNATIVES TECHNOLOGIES ARE EXAMINED USING A RISK-REGULATION STRATEGY TO MEASURE THE ACCEPTABILIT OF EACH TECHNOLOGY.
DS RISK;COMPARISON;RELIABILITY ANALYSIS;N-POWER. SAFETY OF;SAFETY REVIEW;SAFETY EVALUATION;HAZARDS ANALYSIS;SAFETY ANALYSIS REPORT. METEOROLOGY

0000001-00001577 52
ION NO. 00X0166775
SOCIAL COSTS FOR ALTERNATIVE MEANS OF ELECTRIC POWER GENERATION FOR 1980 AND 1990
TH ARGONNE NATIONAL LAB.. IL
1973
A
ANL-8093 (VOL. 4) +. 140 PPS. MARCH 1973
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S.
DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161
RY 230000
N 0130
ODE CZA
Y A
CT THE ENERGY SOURCES CONSIDERED AS ALTERNATES ARE THOSE IN LARGE SCALE USE NOW AS WELL AS THOSE IN ADVANCED STAGES OF DEVELOPMENT. THE AVERAGE INJURY RATES. HEALTH EFFECTS AND DAMAGES TO SURROUNDINGS ARE DEPENDED AS THE RISKS OR SOCIAL COSTS. THE ASSESSMENT IS REPORTED IN TWO DOCUMENTS: A SUMMARY AND A REFERENCE.
DS RISK;HAZARDS ANALYSIS;SAFETY ANALYSIS;POWER PLANT. NUCLEAR; POWER PLANT. FOSSIL FUEL;COMPARISON;SOCIO/PHILOSOPHICAL CONSIDERATION;HAZARD. RELATIVE

0000001-00001577 53
ION NO. 00X0166773
INDEX OF RISK EXPOSURE AND RISK ACCEPTANCE CRITERIA
(S) MILLER S;HALL RE
TH BROOKHAVEN NATIONAL LAB.. UPTON. NY
1981
N
NUREG/CR-1930 + BNL-NUREG-51339 +. 102 PPS. FEB. 1981
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S.
DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161
RY 230000
N 0130
ODE DZA
Y A
CT THIS REPORT CONTAINS ABSTRACTS OF ARTICLES WHICH ADDRESS VARIOUS ASPECTS OF QUANTITATIVE RISK ANALYSIS. COVERING THE SUBJECTS OF NUMERICAL RISK CRITERIA. METHODOLOGY AND DATA. THE ARTICLES WERE PUBLISHED PRIMARILY FROM 1976 TO DATE IN TECHNICAL JOURNALS. REPORTS. PAPERS. ETC. THE ARTICLES SELECTED FOR THIS REPORT WERE JUDGED TO HAVE RELEVANCE TO NUCLEAR POWER PLANT RISK EVALUATIONS AND TO THE POSSIBLE ESTABLISHMENT OF THE ACCEPTABILITY (OR UNACCEPTABILITY) OF CALCULATED NUCLEAR POWER PLANT RISKS. A MATRIX OF THE VARIOUS RISK CRITERIA PROPOSED IS PRESENTED.
DS RISK;HAZARDS ANALYSIS;POPULATION EXPOSURE;ACCIDENT ANALYSIS; HJCK;NRC-RC;SOCIO/PHILOSOPHICAL CONSIDERATION;ACCIDENT. PROBABILITY OF;POWER PLANT. NUCLEAR

0000001-00001577 54
ION NO. 00V0166771
NUCLEAR FOUND NO RISKIER THAN OTHER POWER SOURCES

HOR(S) FISHLOCK D
E 1981
 A
 1 PG. THE ENERGY DAILY, PG. 4 (JAN. 8, 1981)
EGORY 230000;170000
TION 0130
NTRY A
TRACT THE UNITED KINGDOM HEALTH AND SAFETY EXECUTIVE HAVE REACHED THE CONCLUSION THAT NUCLEAR ENERGY IS NO MORE RISKY THAN THE CONVENTIONAL SOURCES OF ELECTRICAL POWER PRODUCTION. NEWS ARTICLE WITH ONE REFERENCE.
WORDS RISK;COMPARISON;POWER PLANT, NUCLEAR;POWER PLANT, FOSSIL FUEL; UNITED KINGDOM

)/0000001-00001577 55
ESSION NO. 00X0166766
LE LONGEVITY BENEFITS AND COSTS OF REDUCING VARIOUS RISKS
HOR(S) SCHWING RC
PAUTH GENERAL MOTORS RESEARCH LABS.
E 1979
 N
 13 PPS, TECHNOLOGICAL FORECASTING & SOCIAL CHANGE, VOL. 13, PP. 333-45 (1979)
EGORY 230000
TION 0130
NTRY A
TRACT INCREASED LONGEVITY IS ONE ALTERNATIVE TO "LIVES SAVED" AS A MEASURE OF BENEFITS DERIVED FROM LARGE-SCALE RISK-REDUCTION PROGRAMS THAT DEMAND RESOURCES OF THE TOTAL POPULATION. THIS MEASURE, FOR SEVERAL CATEGORIES OF RISK, IS PRESENTED IN THE CONTEXT OF THE RISKS PREVALENT IN OUR OWN SOCIETY. SOME ESTIMATES OF THE THEORETICAL BENEFITS DUE TO THE SUCCESSFUL REDUCTION OF RISKS IMPOSED BY INDUSTRY ARE PROVIDED FOR PURPOSES OF ILLUSTRATION. COST-EFFECTIVENESS VALUES AND THESE MEASURES OF PROGRAM IMPACT ON LONGEVITY ARE GRAPHICALLY PRESENTED FOR SEVERAL MORTALITY-REDUCING PROGRAMS. THE RESULTS CLEARLY SHOW THAT THE EFFICIENCY AND THE IMPACT OF POLICIES CAN VARY SO SIGNIFICANTLY THAT PRIORITIES SHOULD BE ESTABLISHED PRIOR TO THEIR IMPLEMENTATION. SINCE THE MATERIAL HAS BEEN DERIVED FROM A VARIETY OF STUDIES IN SEVERAL SECTORS UTILIZING DIFFERENT METHODOLOGIES, THE ACCURACY OF SOME NUMBERS CAN BE QUESTION. THE PURPOSE IS TO PROPOSE THIS WORK AS A PERSPECTIVE FOR POLICY DECISIONS; SINCE LARGE UNCERTAINTIES REMAIN, IT IS NOT TO SUPPLY A DATA BASE FOR DECISIONS.
WORDS RISK;HAZARDS ANALYSIS;SOCIO/PHILOSOPHICAL CONSIDERATION;COST BENEFIT;SAFETY ANALYSIS;MORTALITY

)/0000001-00001577 56
ESSION NO. 0030166766
LE RISK ASSESSMENT FOR RADIATION PROTECTION PURPOSES
HOR(S) PACHIN EE
PAUTH NATIONAL RADIOLOGICAL PROTECTION BOARD, HARWELL, N.K.
E 1980
 C
 24 PPS, 89 REFS, ATOMIC ENERGY REVIEW, 18(3), PP. 779-804
EGORY 230000;150000
TION 0130
NTRY A
 AERE
TRACT IN DEFINING CRITERIA FOR GOOD PROTECTION AGAINST IONIZING RADIATION, IT IS IMPORTANT TO ASSESS QUANTITATIVELY THE LIKELY RISK OF ANY RADIATION EXPOSURE. THE 'SOMATIC' RISKS TO THE INDIVIDUAL RESULT MAINLY FROM INDUCTION OF CANCER IN THE ORGANS IRRADIATED, AND THESE RISKS CAN NOW BE ESTIMATED ON THE BASIS OF NUMEROUS DETAILED EPIDEMIOLOGICAL SURVEY OF EXPOSED HUMAN POPULATIONS. ESTIMATES OF THE RISK OF HEREDITARY EFFECTS, FROM GENETIC CHANGES INDUCED IN GERM CELLS, ARE BASED LARGELY ON THE FREQUENCY WITH WHICH SUCH EFFECTS ARE INDUCED IN OTHER SPECIES. IN BOTH CASES THE RISK AT VERY LOW DOSE CAN BE INFERRED USING KNOWLEDGE OF THE WAY IN WHICH RADIATION DAMAGE IS CAUSED IN TISSUES. COHERENT SYSTEMS OF RADIATION PROTECTION ARE BASED ON A RESTRICTION OF DOSES TO THE WHOLE BODY AND TO INDIVIDUAL ORGANS, SUCH THAT THE INDUCTION OF CANCER AND GENETIC HARM IS INFREQUENT, AND THE THRESHOLD DOSE FOR CAUSING OTHER,

'NON-STOCHASTIC', EFFECTS IS NOT EXCEEDED.
DS   RISK;RADIATION SAFETY AND CONTROL;RADIATION EXPOSURE;RADIATION EFFECT, SPECIES;SOCIO/PHILOSOPHICAL CONSIDERATION;HAZARDS ANALYSIS;DOSE;CANCER;EFFECT, SOMATIC

/0000001-00001157/   57
SSION NO.   00X0166760
E   GERMAN RISK STUDY-MAIN REPORT-A STUDY OF THE RISKS DUE TO ACCIDENTS IN NUCLEAR POWER PLANTS
OR(S)   BARSELL AW;WALL IB
AUTH   ELECTRIC POWER RESEARCH INST., PALO ALTO, CA
     1981
     N
     EPRI-NP-1804-SR +. 323 PPS, FIGS, APRIL 1981
L   AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10090, PALO ALTO, CALIF. 94303
GORY   230000
ION   0130
 CODE   EPR
TRY   A
RACT   A TRANSLATION OF THE "DEUTSCHE RISIKOSTUDIE KERNKRAFTWERKE." THIS REPORT ASSESSES THE RISKS DUE TO ACCIDENTS CAUSED BY THE OPERATION OF NUCLEAR POWER PLANTS IN THE FEDERAL REPUBLIC OF GERMANY. THE STUDY, PERFORMED UNDER THE DIRECTION OF THE FEDERAL MINISTRY FOR RESEARCH AND TECHNOLOGY, IS ORGANIZED INTO TWO PHASES: THE CURRENT REPORT PRESENTS AN OVERVIEW OF THE INVESTIGATIONS AND RESULTS OF THE FIRST PHASE (PHASE A), WHEREBY THE PROBABILISTIC RISK ASSESSMENT METHODOLOGY USED IN THE AMERICAN REACTOR SAFETY STUDY (WASH-1400) IS APPLIED TO THE PARTICULAR REACTOR SYSTEM TECHNOLOGY AND SITING CONDITIONS OF THE FRG. WITHIN THE UNCERTAINTIES, THE RESULTS OF THE GERMAN AND AMERICAN STUDIES AGREE, THE GERMAN STUDY CONFIRMING THAT ACCIDENT-CAUSED RISKS FROM NUCLEAR POWER PLANTS ARE RELATIVELY SMALL.
ORDS   SOCIO/PHILOSOPHICAL CONSIDERATION;RISK;ACCIDENT ANALYSIS; ACCIDENT, CONSEQUENCES;HAZARDS ANALYSIS;FAILURE MODE ANALYSIS; NUCLEAR DEBATE;ACCIDENT, PROBABILITY OF

0/0000001-00001157/   58
SSION NO.   00X0166759
E   SPASM, A COMPUTER CODE FOR MONTE CARLO SYSTEM EVALUATION
OR(S)   LEVERENZ FL
AUTH   SCIENCE APPLICATIONS INC., PALO ALTO, CA
     1981
     N
     EPRI-NP-1685 +. 31 PPS, 5 TABS, 9 FIGS, JAN. 1981
L   AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10090, PALO ALTO, CALIF. 94303
GORY   230000
ION   0130
 CODE   SAI
TRY   A
RACT   SPASM IS A COMPUTER CODE IN THE WAM FAMILY WHICH EVALUATES THE DISTRIBUTION OF SYSTEM SUCCESS OR FAILURE FREQUENCY VIA MONTE CARLO SIMULATION. IT CAN BE USED INDEPENDENTLY OR IN CONJUNCTION WITH WAMCUT.
ORDS   COMPUTER PROGRAM;STATISTICAL ANALYSIS;RISK;FAULT TREE ANALYSIS; EPRI

0/0000001-00001157/   59
SSION NO.   00X0166757
E   STATUS REPORT ON EPRI FUEL CYCLE ACCIDENT RISK ASSESSMENT
OR(S)   ERDMANN RC;GARCIA AA;STEVENS CA
AUTH   SCIENCE APPLICATIONS INC., PALO ALTO, CA
     1979
     N
     EPRI-NP-1128 +. 182 PPS, JULY 1979
L   AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10090, PALO ALTO, CALIF. 94303
GORY   230000;130000
ION   0130
 CODE   SAI
TRY   A
RACT   PRESENTS THE STATUS OF THE CURRENT EPRI PROJECT FOR ASSESSING

THE RISKS OF PROCESSES THAT SUPPORT NUCLEAR POWER PLANTS. IT SUMMARIZES AND EXTENDS THE WORK REPORTED IN FIVE UNPUBLISHED DRAFT REPORTS. FOUR OF THESE CONSIDER THE ACCIDENTAL RADIOLOGICAL RISK OF REPROCESSING SPENT FUEL, MIXED OXIDE FUEL FABRICATION, THE TRANSPORTATION OF MATERIALS WITHIN THE FUEL CYCLE, AND THE DISPOSAL OF NUCLEAR WASTES. THE OTHER REPORT IS CONCERNED WITH THE ROUTINE ATMOSPHERIC RADIOLOGICAL RISK OF MINING AND MILLING URANIUM-BEARING ORE. THE PRESENT RESULTS SHOW THAT THE TOTAL RISK CONTRIBUTION OF THE FUEL CYCLE IS ONLY ABOUT 1% OF THE ACCIDENT RISK OF THE POWER PLANT AND HENCE, WITH LITTLE ERROR, THE ACCIDENT RISK OF NUCLEAR ELECTRIC POWER IS THAT OF THE POWER PLANT ITSELF. THE POWER PLANT RISK, ASSUMING A VERY LARGE USAGE OF NUCLEAR POWER BY THE YEAR 2005, IS ONLY ABOUT 0.5% OF THE RADIOLOGICAL RISK OF NATURAL BACKGROUND.

YWORDS RISK;FUEL CYCLE;SAFETY PRINCIPLES AND PHILOSOPHY;HAZARDS ANALYSIS;HAZARD, RELATIVE;EPRI;ACCIDENT, PROBABILITY OF

//0/0000001-0000115// 60

CESSION NO. 00X0166756

ILE TENTATIVE GUILINE OF WORK SCOPE AND ORGANIZATION FOR A STUDY OF THE RISKS FROM ENERGY PRODUCTION IN THE U.S.

THOR(S) CAVE L;OKRENT D;PERY G

RPAUTH UNIV. OF CALIF., LOS ANGELES

TE 1979

PE N

MO ALO-81 + UCLA-ENG-7941 +. 37 PPS, FIGS, REFS, JULY 1979

AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

TEGORY 230000

ITION 0130

RP CODE UAV

UNTRY A

STRACT AS A RESULT OF INCREASING AWARENESS OF THE HAZARDS TO HEALTH AND SAFETY ASSOCIATED WITH ENERGY TECHNOLOGIES, THE STUDY OF RISKS HAS BECOME AN ESSENTIAL FEATURE OF THE DECISION MAKING PROCESS. THEREFORE IT IS PROPOSED TO ESTABLISH AN ORGANIZATION TO STUDY ALL RISKS ASSOCIATED WITH ENERGY PRODUCTION. AN OUTLINE OF METHODOLOGY AND STRUCTURE OF THE NECESSARY ORGANIZATION IS GIVEN.

YWORDS RISK;ENERGY POLICY;ORGANIZATION, INTERNATIONAL;PRODUCTION

/0/0000001-0000115// 61

ESSION NO. 00X0166754

LE ON THE DEVELOPMENT OF QUANTITATIVE RISK ACCEPTANCE CRITERIA

HOR(S) OKRENT D;GRIESMERGER JM

PAUTH UNIV. OF CALIF., LOS ANGELES

E 1981

E N

O ALO-130 + UCLA-ENG-7969 +. 45 PPS, FIGS, REFS, JAN. 1981

IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

EGORY 230000

TION 0130

P CODE UAV

NTRY A

TRACT MANAGEMENT OF RISKS IS AS MUCH A SOCIO-POLITICAL PROBLEM AS A TECHNICAL ONE. SOCIETY IS BECOMING INCREASINGLY AWARE OF THE FACTS THAT RISKS ACCOMPANY THE BENEFITS AND OTHER COSTS OF ITS TECHNOLOGICAL VENTURES. THESE RISKS CANNOT BE TOTALLY ELIMINATED: THEY CAN ONLY BE REDUCED SO THAT THEY ARE ONLY ONE OF THE ELEMENTS IN THE DECISION PROCESS. SOME OF THE MAJOR CONSIDERATIONS FOR EFFECTIVE MANAGEMENT OF RISKS ARE DISCUSSED, WITH EMPHASIS ON RISKS DUE NUCLEAR POWER AS SHOWN IN THIS PAPER.

WORDS RISK;HAZARDS ANALYSIS;SOCIO/PHILOSOPHICAL CONSIDERATION; DECISION ANALYSIS;NUCLEAR DEBATE;ANALYTICAL MODEL;COMPARISON

/0/0000001-0000115// 62

SSION NO. 00X0166753

LE RISK MANAGEMENT AND DECISION RULES FOR LIGHT WATER REACTORS

HOR(S) OKRENT D;GRIESMEYER JM

AUTH UNIV. OF CALIF., LOS ANGELES

1981

N
ALO-134 + UCLA-ENG-8054 +. 27 PPS. TABS. JAN. 1981
L AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD, VA. 22161
EGORY 230000
TION 0130
P CODE UAV
NTRY A
STRACT A CENTRAL ISSUE IN ENERGY POLICY IS THE CONDUCTION OF THE RISKS FROM VARIOUS TECHNOLOGIES THAT ARE USED TO GENERATE ELECTRIC POWER. PART OF THE CONFLICT IS IN THE ATTEMPT TO BALANCE RISKS AND BENEFITS END IS DEEPLY ROOTED IN THE SOCIETAL VALUES USED IN MAKING COMPARISONS. THIS SITUATION IS FURTHER COMPLICATED BY THE LARGE UNCERTAINTIES ENRICHED IN THE ESTIMATIONS. A PRELIMINARY PROPOSAL FOR LIGHT WATER REACTOR RISK MANAGEMENT FRAMEWORK IS PRESENTED ALONG WITH A SET OF DECISION MAKING RULES.
YWORDS HAZARDS ANALYSIS;RISK;COMPARISON;SAFETY EVALUATION;SAFETY ANALYSIS;REACTOR, LWR

/70/0000001-00001157/ 63
CESSION NO. 00X0166752
TLE ON RISK AVERSION IN RISK ACCEPTANCE CRITERIA
THOR(S) WU-CHIEN JS;APOSTOLAKIS G
RPAUTH UNIV. OF CALIF., LOS ANGELES
TE 1981
PE N
MO ALO-136 + UCLA-ENG-8062 +. 15 PPS. 2 FIGS. 10 REFS. MARCH 1981
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD, VA. 22161
TEGORY 230000
TION 0130
P CODE UAV
NTRY A
TRACT THE RISK AVERSE ATTITUDE THAT IS INCLUDED IN SOME PROPOSED RISK ACCEPTANCE CRITERIA IS EXAMINED. IT IS SHOWN THAT IT IS A WEAKER ATTITUDE THAN RISK AVERSION, AS IS COMMONLY DEFINED IN DECISION THEORY. CONSEQUENTLY, THE BOUNDARY CURVE SEPARATING ACCEPTABLE AND UNACCEPTABLE REGIONS DOES NOT HAVE TO BE A STRAIGHT LINE ON THE LOGARITHMIC FREQUENCY-CONSEQUENCE SPACE. A CURVE OF VARIABLE SLOPE WOULD EXPRESS THE SAME ATTITUDE AS LONG AS THE SLOPE IS LESS THAN -1.
WORDS RISK;SOCIO/PHILOSOPHICAL CONSIDERATION;SAFETY ANALYSIS;DECISION ANALYSIS;ANALYTICAL MODEL;COMPARISON;HAZARDS ANALYSIS

/70/0000001-00001157/ 64
ESSION NO. 00X0168357
LE REVIEW OF PROPOSED IMPROVEMENTS, INCLUDING FILTER/VENT OF BWR PRESSURE-SUPPRESSION AND PWR ICE CONTAINMENTS
HOR(S) LEVY S;GERBER TL
PAUTH S. LEVY INC., CAMPBELL, CA
E 1981
E N
O EPRI-NP-1747 +. 215 PPS. TABS. FIGS. REFS. APRIL 1981
IL AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10090, PALO ALTO, CALIF. 94303
EGORY 120000;110000;230000;050000
TION 0129
NTRY A
TRACT A STATE OF THE ART REVIEW OF IMPROVEMENTS INCLUDING FILTER/VENT OF BWR PRESSURE SUPPRESSION AND PWR ICEN CONTAINMENTS HAS BEEN CARRIED OUT. IT INCLUDES A SUMMARY DESCRIPTION OF OPERATING AND PLANNED BWR PRESSURE SUPPRESSION AND PWR ICE CONTAINMENTS; A REVIEW OF IMPROVEMENTS PROPOSED TO-DATE; AN ASSESSMENT AND AN UPDATING OF THE RISKS ASSOCIATED WITH THE VARIOUS CONTAINMENT FAILURE MODES AS DEVELOPED IN THE REACTOR SAFETY STUDY; THE FORMULATION OF A DESIGN IMPROVEMENT STRATEGY BASED UPON THE DOMINANT RISK SCENARIOS; AN EVALUATION OF VARIOUS POTENTIAL PREVENTIVE AND MITIGATION CONCEPTS AND, IN PARTICULAR, THE APPLICATION OF FILTER/VENT SYSTEMS; AND FINALLY, A SET OF RECOMMENDATIONS.
ORDS REACTOR, BWR;CONTAINMENT, ICE CONDENSER;REACTOR, PWR;SAFETY REVIEW;CONTAINMENT, PRESSURE SUPPRESSION;CONTAINMENT FILTERING SYSTEM;RISK;EPRI;SAFETY REVIEW;FAILURE MODE ANALYSIS;

CONTAINMENT FILTERING SYSTEM;SAFETY REVIEW;FAILURE MODE ANALYSIS,CONTAINMENT FILTERING SYSTEM;SAFETY REVIEW;FAILURE MODE ANALYSIS;CONTAINMENT FILTERING SYSTEM

/70/0000001-00001157/ 65
ESSION NO. 00J0166278
TLE DISTINCTIVE ASPECTS OF THE HTGR
TE 1981
PE O
MO 4 PPS, NUCLEAR ENGINEERING INTERNATIONAL, 26(313), PP. 22-25 (MAY 1981)
TEGORY 170000;120000;230000
ITION 0129
UNTRY O
ABB NEIN
STRACT THE MANY INHERENT AND ENGINEERED SAFETY FEATURES OF THE HTGR WHICH CAUSE RELEASES FROM A WIDE SPECTRUM OF SEVERE ACCIDENTS TO BE LOW ARE EXAMINED. METHODS USING PROBABILISTIC RISK ASSESSMENT HAVE MORE RECENTLY BEEN USED TO JUDGE THE VALUE OF FURTHER OPTIONS, SUCH AS THE ADDITION OF NATURAL CONVECTION COOLANT LOOPS.
YWORDS REACTOR, HTGR;FT. ST. VRAIN (HTGR);CONCRETE, PRESTRESSED; ENGINEERED SAFETY FEATURE;DECAY HEAT;AUXILIARY COOLING; OPTIMIZATION;RISK;TRANSIENT

7/07/0000001-00001157/ 66
CESSION NO. 00J0166262
TLE REALISTIC ESTIMATES OF THE CONSEQUENCES OF NUCLEAR ACCIDENTS
THOR(S) LEVENSON M;RAHN F
RPAUTH ELECTRIC POWER RESEARCH INST., PALO ALTO, CA
TE 1981
PE O
MO 11 PP, 3 TABS, 46 REFS, NUCLEAR TECHNOLOGY, 53(2), PP. 99-110 (MAY 1981)
TEGORY 230000;140000
ITION 0129
RP CODE EPR
UNTRY A
ABB NUAT
STRACT THE PRINCIPAL AREAS OF CONCERN FOCUS ON THE TREATMENT OF A NUMBER OF PHYSICAL PROCESSES. THESE PROCESSES ARE ALWAYS OPERATIVE AND CAN BE COUNTED ON TO LIMIT THE CONSEQUENCES OF A REACTOR ACCIDENT. SUFFICIENT CREDIT IS NOT TAKEN FOR THEIR ABILITY TO REDUCE THE RELEASE OF RADIOACTIVITY AND CONFINE IT RELATIVELY CLOSE TO ITS SOURCE. ESTIMATES OF RISK WILL IMPROVE IN DIRECT PROPORTION TO IMPROVEMENTS IN QUANTIFICATION OF THESE PHENOMENA. EMPIRICAL EVIDENCE FROM MANY SOURCES SHOWS THAT THESE PROCESSES ARE INDEED OPERATIVE AND VERY EFFICIENT IN REDUCING THE RELEASE OF RADIOACTIVITY. AS A RESULT, THE POLICY DECISIONS BASED ON THE SOURCE TERM IN THE EVENT OF A MAJOR REACTOR ACCIDENT MUST BE REASSESSED.
WORDS ACCIDENT;ACCIDENT, PROBABILITY OF;RISK;RADIOACTIVITY RELEASE; ACCIDENT, CONSEQUENCES;CORE MELTDOWN;DAMAGE;FIRE

/70/0000001-00001157/ 67
ESSION NO. 00J0166165
LE TMI CLEANUP LESS RISKY THAN CURRENT CONDITION, SAYS NRC STAFF
E 1981
E O
O 2 PPS, NUCLEAR INDUSTRY, 28(4), PP. 20-21 (APRIL 1981)
EGORY 170000;150000;230000
TION 0129
NTRY A
BB NUID
TRACT THE PROGRAMATIC ENVIRONMENTAL IMPACT STATEMENT QUANTIFIES THE ESTIMATED RISKS TO AN "INDIVIDUAL OFFSITE RECEIVING THE MAXIMUM ESTIMATED DOSE" AS AN INCREASED RISK OF CANCER DEATH FROM BETWEEN ONE IN TWO MILLION TO ONE IN 700,000. THE NRC STAFF EMPHASIZED THE NEED "TO PROCEED AS EXPEDITIOUSLY AS IS REASONABLE FEASIBLE" IN THE CLEANUP.
WORDS THREE MILE ISLAND 2 (PWR);REACTOR, PWR;INCIDENT;DECONTAMINATION; AGENCY, NRC;STATEMENT, ENVIRONMENTAL;RISK;CANCER;DOSE;INCIDENT, RECOVERY FROM

SSION NO. 00C0166006
L NONSAFETY LOADS ON CLASS 1E POWER SOURCES, REVISITED
HOR(S) LEWIN J
PAUTH OAK RIDGE NATIONAL LAB., TN
E 1981
E L
D 3 PPS, IEEE TRANS. NUCL. SCI., 28(1), PP. 929-31 (FEB. 1981)
TEGORY 100000;090000;120000;230000
ITION 0128
RP CODE F2C
UNTRY A
STRACT NONSAFETY LOADS ON CLASS 1E POWER SOURCES, WITHIN LIMITS OF PRUDENT ENGINEERING PRACTICE AND WITH FAILURE RATES CHARACTERISTIC OF CURRENT EQUIPMENT, DO NOT CONSTITUTE A DISTINGUISHABLE RISK INCREMENT FOR THE AVAILABILITY OF EMERGENCY ELECTRICAL POWER AT NUCLEAR PLANTS.
YWORDS ENGINEERED SAFETY FEATURE;EMERGENCY POWER, ELECTRIC;ACCIDENT; RATE;FAULT TREE ANALYSIS;FAILURE, EQUIPMENT;RISK

7/0/0000001-0000115// 69
CESSION NO. 00E0165751
ILE SAFEGUARDS ANALYSIS OF A SPENT LWR FUEL RECYCLE COMPLEX
THOR(S) HEINRICH LA
RPAUTH SAVANNAH RIVER LAB., AIKEN, S.C.
TE 1980
PE H
MD DP-1547 +. 66 PPS, 8 TABS, 19 FIGS, 19 REFS, AUG. 1980
AIL AVAILABILITY - LIMITATIONS ON DISTRIBUTION; SEND REQUESTS TO DOE TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830
TEGORY 220000;130000;230000
ITION 0128
RP CODE SRL
UNTRY A
STRACT A SAFEGUARDS ANALYSIS WAS MADE OF TWO OPTIONS FOR REPROCESSING IRRADIATED LIGHT-WATER REACTOR (LWR) FUEL. ONE OPTION IS TO COMPLETELY PARTITION PLUTONIUM FROM URANIUM AND FISSION PRODUCTS; THE OTHER OPTION IS TO COPROCESS PLUTONIUM WITH PART OF THE URANIUM IN A PRODUCT STREAM NEVER CONTAINING MORE THAN TEN PERCENT PLUTONIUM. EACH OPTION WAS DEVELOPED TO A DEGREE THAT INCLUDED A CONCEPTUAL DESIGN, WITH ESTIMATED CAPITAL COSTS FOR ALL FACILITIES NECESSARY TO REPROCESS THE IRRADIATED LWR FUEL AND PRODUCE URANIUM-PLUTONIUM MIXED OXIDE (MOX) FUEL FOR REIRRADIATION.
WORDS SAFEGUARDS, NUCLEAR MATERIAL;FUEL REPROCESSING;THEFT/DIVERSION; SOLVENT EXTRACTION PROCESS;PARTITION COEFFICIENT;WASTE TREATMENT;REACTOR, LWR;PLUTONIUM;SABOTAGE;TRITIUM;COST ANALYSIS; RISK

7/0/0000001-0000115// 70
ESSION NO. 00X0165347
LE UFOMOD PROGRAM TO CALCULATE THE RADIOLOGICAL CONSEQUENCES OF REACTOR ACCIDENTS WITHIN RISK STUDIES (IN GERMAN)
HOR(S) SCHUCKLER M;VOGT S
PAUTH KERNFORSCHUNGSZENTRUM KARLSRUHE, F. R. GERMANY
E 1981
E N
D KFK-3092 + GERRSR-706 +. 160 PPS, FIGS, JAN. 1981
GUAGE GERMAN
IL AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
EGORY 150000;180000;190000;230000
TION 0127
P CODE KSR
NTRY G
RACT THE FORTRAN-IV COMPUTER CODE UFOMOD CALCULATES THE RADIOLOGICAL CONSEQUENCES OF REACTOR ACCIDENTS FOR RISK STUDIES, NAMELY EARLY DEATHS, LATENT CANCER DEATHS AND GENETIC SIGNIFICANT DOSES. DIFFERENT MODELS FOR THE ATMOSPHERIC TRANSPORT AND DEPOSITION, THE DOSE CALCULATION, THE COUNTERMEASURES AND THE INJURIES ARE USED TO CALCULATE INDIVIDUAL AND COLLECTIVE INJURY. UP TO 54 RADIONUCLIDES, 10 RELEASE CATEGORIES, 4 METEOROLOGICAL ZONES, 10 POPULATION DISTRIBUTIONS PER ZONE WITH

UP TO 36 SECTORS AND 50 RINGS, AND 115 WEATHER SEQUENCES PER ZONE MAY BE USED. THE DETERMINISTIC RESULTS ARE COMBINED TOGETHER WITH THE RESPECTIVE PROBABILITIES AND FREQUENCIES TO GIVE COMPLEMENTARY CUMULATIVE FREQUENCY DISTRIBUTIONS. THIS REPORT DESCRIBES THE COMPUTER CODE AND ITS INPUT AND OUTPUT.
ORDS GERMANY;COMPUTER PROGRAM;ACCIDENT;EFFECT, GENETIC;DOSE; ATMOSPHERIC DIFFUSION;DEPOSITION;ACCIDENT, CONSEQUENCES;FOREIGN EXCHANGE;RISK;POPULATION EXPOSURE;MORTALITY;RADIOACTIVITY RELEASE

/0000001-00001577 71
SSION NO. 00J0165135
E COMPROMISE ON BEIR III
1980
O
2 PPS. NUCLEAR NEWS, 23(11), PP. 69-70 (SEPT. 1980)
GORY 190000;150000;230000
ION 0126
TRY A
NONE
RACT A COMPROMISE POSITION ON THE CONTROVERSIAL ISSUE OF THE POSTULATED EFFECTS OF VERY LOW DOSES OF LOW-LEVEL RADIATION HAS PERMITTED THE NATIONAL ACADEMY OF SCIENCES TO RELEASE BEIR III - THE THIRD RENDERING OF ITS COMMITTEE ON THE BIOLOGICAL EFFECTS OF IONIZING RADIATION. THE FINAL BEIR III REPORT SAYS THE METHODOLOGY USED FOR CALCULATING THE BALANCED-VIEW RANGE OF CANCER RISKS CONSIDERED A FAMILY OF THEORETICAL DOSE-RESPONSE MODELS FOR ESTIMATING THE RELATIONSHIP BETWEEN RADIATION DOSE AND CANCER CASES. IN THESE BALANCED-VIEW STUDIES, EMPHASIS WAS PLACED ON THE "LINEAR-QUADRATIC" MODEL, WHICH PROVIDES A MEDIAN CANCER-RISK ESTIMATE BETWEEN THE UPPER AND LOWER ESTIMATES PROJECTED BY THE PURE LINEAR AND QUADRATIC MODELS. THUS, A COMPROMISE POSITION WAS STRUCK BY THE BALANCED-VIEW PANEL.
ORDS LOW LEVEL RADIATION;RADIATION EFFECT;RADIATION EXPOSURE, CHRONIC;RADIATION SAFETY AND CONTROL;RADIATION PROTECTION, ORGANIZATION;RISK;CANCER

/0000001-00001577 72
SSION NO. 00X0165020
E CERTAIN APPROACHES TO RADIATION FACTOR STANDARDIZATION (IN RUSSIAN)
OR(S) TERMAN AV;MOISEER AA
1978
N
6 PPS. IZVT. SSSR. NO. 52-53, PP. 88-91 (1978)
UAGE OTHER LANG
GORY 150000;230000
ION 0126
TRY R
RACT SOME PROCEDURAL ASPECTS IN THE EVALUATION OF THE ACCEPTABLE RISK FROM A RADIATION FACTOR ARE CONSIDERED. THE USE OF ONCOLOGIC MORBIDITY OR MORTALITY, WHICH ARE AVERAGED FOR ALL THE POPULATION AS A WHOLE, PROVED TO BE UNRELIABLE AS A CRITERIUM FOR EVALUATION OF ACCEPTABLE RADIATION RISK. IT IS SUGGESTED THAT A RELATIVE IRRADIATION RISK SHOULD BE DETERMINED BY COMPARING THE INCIDENCE OF IRRADIATION-INDUCED NEOPLASMS WITH THE SPONTANEOUS FREQUENCY OF THESE NEOPLASMS IN THE AGE GROUP OF UP TO 30 YEARS OLD. AN EQUATION IS GIVEN FOR CALCULATING A MAXIMUM PERMISSIBLE IRRADIATION LIMIT FOR LARGE POPULATION GROUPS.
ORDS EFFECT, DOSE;RADIATION EFFECT;RADIATION EXPOSURE;POPULATION EXPOSURE;DOSE CALCULATION, EXTERNAL;DOSE CALCULATION, INTERNAL; USSR;RISK;PUBLIC RELATIONS

/0000001-00001577 73
SSION NO. 00J0165016
E PLANT SAFETY AND PUBLIC RISK IN NUCLEAR ENERGY
OR(S) DEUTSCH RW
AUTH GENERAL PHYSICS CORP.
1980
O
4 PPS. PUBLIC UTILITIES FORTNIGHTLY, 106(11), PP. 18-21 (NOV. 20, 1980)
GORY 010000;230000

ION 0126
TRY A
B PUPN
TRACT "NO COMPLEX FACILITY DESIGNED AND OPERATED BY MAN CAN BE ABSOLUTELY SAFE," SAYS THE AUTHOR - NOT EVEN A NUCLEAR REACTOR UTILIZED FOR ELECTRIC POWER GENERATION. "THERE IS A VERY SMALL PROBABILITY THAT A SERIES OF HIGHLY UNLIKELY EVENTS COULD OCCUR SIMULTANEOUSLY, RESULTING IN A FAILURE OF THE CONTAINMENT STRUCTURE WITH THE RELEASE OF LARGE QUANTITIES OF RADIOACTIVITY TO THE ATMOSPHERE." HIS ARTICLE PROVIDES AN UNUSUALLY SOBERHEADED APPRAISAL OF THE PROSPECTS FOR NUCLEAR MISHAP AND THE HUMAN CONSEQUENCES - INJURY AND DEATH TO PERSONS AND LOSS OF PROPERTY AND MONEY. IT ALSO TELLS WHAT CAN BE DONE, AND IS BEING DONE, TO MINIMIZE THE PROBABILITIES OF NUCLEAR ACCIDENT AND ATTENDANT LOSSES. (EWH)
YWORDS SECURITY;N-POWER, SAFETY OF;RADIATION SAFETY AND CONTROL;RISK; BENEFIT VS RISK;ACCIDENT ANALYSIS;SAFETY PRINCIPLES AND PHILOSOPHY;SOCIO/PHILOSOPHICAL CONSIDERATION

//0/0000001-00001157/ 74
CESSION NO. 0000164974
TLE SPECIFICATION OF COMPUTATIONAL APPROACH
THOR(S) WALL IH;KAUL MK;POST RI
RPAUTH LAWRENCE LIVERMORE LAB., CALIF.
TE 1981
PE U
MD NUREG/CR-1702 + UCRL-15985 +. 157 PPS. 3 TABS. 19 FIGS. JAN. 1981
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 020000;230000
ITION 0126
RP CODE L2C
UNTRY A
STRACT THE WORK REPORTED IN THIS DOCUMENT WAS PERFORMED UNDER SUBCONTRACT TO THE LAWRENCE LIVERMORE LABORATORY AS PART OF THE SEISMIC SAFETY MARGINS RESEARCH PROGRAM (SSMRP) SPONSORED BY THE U.S. NUCLEAR REGULATORY COMMISSION. TWO SUBCONTRACTORS WERE ASKED TO DEVELOP SPECIFICATIONS FOR A COMPUTATIONAL APPROACH TO BE USED TO HELP ACCOMPLISH THE OBJECTIVES OF THE SSMRP. THIS DOCUMENT REPRESENTS THE SPECIFICATIONS DEVELOPED BY ONE OF THE SUBCONTRACTORS. THE WORK WAS PERFORMED IN CLOSE COORDINATION WITH LLL BUT REPRESENTS THE APPROACH SUGGESTED BY THE SUBCONTRACTOR. WHAT THE ACTUAL COMPUTATIONAL APPROACH WILL BE HAS YET TO BE DETERMINED AND INFORMATION DEVELOPED IN THIS SUBCONTRACT WILL BE USED TO HELP DETERMINE THAT APPROACH.
WORDS SITING;SEISMIC DESIGN;SEISMOLOGY;INTERACTION, SOIL AND STRUCTURE;HAZARDS ANALYSIS;STATISTICAL ANALYSIS;SENSITIVITY ANALYSIS;ANALYTICAL MODEL;MATHEMATICAL STUDY;RISK;HJCK;NRC-RD; NRC-RM

/0/0000001-00001157/ 75
ESSION NO. 0000164925
LE NRC STAFF ISSUES THREE REPORTS ON URBAN TRANSPORT OF RADIOACTIVE MATERIALS
PAUTH U.S. NUCLEAR REGULATORY COMMISSION
E 1980
E M
D NRC NEWS RELEASE 80-165 +. 1 PG. FOR WEEK ENDING SEPT. 23, 1980
EGORY 030000;220000;230000
TION 0126
P CODE NRC
NTRY A
RACT THE PRINCIPAL ONE IS "TRANSPORTATION OF RADIONUCLIDES IN URBAN ENVIRONS: DRAFT ENVIRONMENTAL ASSESSMENT" (NUREG/CR-0743). EMPHASIS IS PLACED ON ESTIMATING DIRECT RADIOLOGICAL IMPACTS SUCH AS HEALTH AND ECONOMIC RISKS AND CONSEQUENCES OF SEVERE INCIDENTS. OTHER REPORTS WHICH WILL BE PART OF THE TECHNICAL BASE FOR THE STAFF'S STUDY ARE "IDENTIFICATION AND ASSESSMENT OF THE SOCIAL IMPACTS OF TRANSPORTATION OF RADIOACTIVE MATERIALS IN URBAN ENVIRONMENTS" (NUREG/CR-0744) AND "REVIEW AND INTEGRATION OF EXISTING LITERATURE CONCERNING POTENTIAL SOCIAL IMPACTS OF TRANSPORTATION OF RADIOACTIVE MATERIALS IN

URBAN AREAS" (NUREG/CR-0742).
ORDS ACENCY, NRC;ENVIRONMENT;RISK;SOCIO/PHILOSOPHICAL CONSIDERATION; TRANSPORTATION AND HANDLING

/0/0000001-00001157/ 76
ESSION NO. 00J0164918
LE IMAGES OF DISASTER: PERCEPTION AND ACCEPTANCE OF RISKS FROM NUCLEAR POWER
THOR(S) SLOVIC P;LICHTENSTEIN S;FISCHHOFF B
TE 1979
PE U
MO 13 PPS, 4 TABS, 4 FIGS, 31 REFS, ELECTRIC PERSPECTIVES, 79/3, PP. 8-20 (JUNE 1979)
TEGORY 010000;230000
ITION 0126
UNTRY A
STRACT ALTHOUGH OPPOSITION TO NUCLEAR POWER HAS MANY CAUSES, CONCERNS ABOUT SAFETY UNDOUBTEDLY PLAY A MAJOR ROLE. OVER THE PAST SEVERAL YEARS, ELECTRIC PERSPECTIVES AND OTHERS HAVE BEEN SYSTEMATICALLY INVESTIGATING PUBLIC PERCEPTIONS OF RISK FROM NUCLEAR POWER AND OTHER HAZARDOUS ACTIVITIES. FROM THIS RESEARCH, A QUANTITATIVE DESCRIPTION OF THE ATTITUDES, PERCEPTIONS AND EXPECTATIONS OF SOME MEMBERS OF THE ANTI-NUCLEAR PUBLIC HAS BEEN PIECED TOGETHER. THE IMAGES OF POTENTIAL NUCLEAR DISASTERS THAT HAVE BEEN FORMED IN THESE PEOPLE'S MINDS ARE REMARKABLY DIFFERENT FROM THE ASSESSMENTS PUT FORTH BY MOST TECHNICAL EXPERTS. (EWH)
YWORDS RISK;BENEFIT VS RISK;SOCIO/PHILOSOPHICAL CONSIDERATION;PUBLIC RELATIONS;RADIATION, PUBLIC EDUCATION;DISASTER;NUCLEAR DEBATE; HAZARD, RELATIVE

//0/0000001-00001157/ 77
ESSION NO. 00J0164916
LE DECEPTION ON NUCLEAR POWER RISKS: A CALL FOR ACTION
THOR(S) WELCH BL
TE 1980
PE U
MO 4 PPS, BULLETIN ATOMIC SCIENTISTS, 36(7), PP. 50-53 (SEPT. 1980)
TEGORY 010000;230000
TION 0126
UNTRY A
ABB SUIA
STRACT SUBSTANTIAL EVIDENCE SUGGESTS THAT FOR A QUARTER OF A CENTURY THE U.S. FEDERAL BUREAUCRACY AND THE NUCLEAR INDUSTRY HAVE DELIBERATELY DECEIVED THE PUBLIC ABOUT THE RISKS OF NUCLEAR POWER. FACTS APPEAR TO HAVE BEEN SYSTEMATICALLY WITHHELD OR DISTORTED AND CALCULATIONS BIASED IN ORDER TO PRESENT NUCLEAR POWER IN A FAVORABLE LIGHT. DISCREPANCIES EXIST BETWEEN WHAT RESPONSIBLE SCIENTISTS TOLD THE PUBLIC AND SOMETIMES THE CONGRESS ABOUT THE RISKS OF NUCLEAR POWER AND WHAT THEY KNEW THE INFORMATION AVAILABLE TO THEM ACTUALLY SUGGESTED.
WORDS SPOKESMAN, FEDERAL;AGENCY, FEDERAL;REGULATION, FEDERAL; INDUSTRY, NUCLEAR;RISK;POWER PLANT, NUCLEAR;HAZARD, RELATIVE; HAZARDS ANALYSIS;SAFETY EVALUATION;SAFETY MARGIN

/0/0000001-00001157/ 78
ESSION NO. 00C0164908
LE NUCLEAR REACTOR SAFETY SESSION
E 1980
E L
O 107 PPS, PP. 297 THRU 403 OF TRANS. OF THE AMERICAN NUCLEAR SOCIETY, VOL. 35, FROM 1980 WINTER MEETING; WASHINGTON, D.C., NOV. 16-21, 1980
IL AVAILABILITY - AMERICAN NUCLEAR SOCIETY PUBLICATIONS, 555 N. KENSINGTON AVE., LA GRANGE PARK, ILL. 60525
EGORY 050000;230000;060000;110000;170000;180000
TION 0126
NTRY A
RACT THESE SESSIONS INCLUDED: THERMAL REACTOR SAFETY (THERMAL-HYDRAULIC AND HEAT TRANSFER PHENOMENOLOGY, REACTOR TRANSIENTS AND ACCIDENTS, AND FUEL ANALYSIS AND EXPERIMENTS) (3 SESSIONS), FAST REACTOR SAFETY (RELATED PHENOMENOLOGY, POST ACCIDENT HEAT REMOVAL, AND EXPERIMENTS AND STRUCTURAL DYNAMICS) (3 SESSIONS), PLUS FAST REACTOR SAFETY ANALYSES, RISK AND

RELIABILITY ANALYSES, AND ACCEPTABLE RISK CRITIERIA AND DECISION MAKING.
WORDS REACTOR, THERMAL;REACTOR, PWR;REACTOR, LWR;REACTOR, BWR; REACTOR, LMFBR;REACTOR, BREEDER;REACTOR, FAST;RISK;PUBLIC RELATIONS;THERMAL HYDRAULIC ANALYSIS;ACCIDENT, HYPOTHETICAL; CORE MELTDOWN;STRUCTURAL ANALYSIS, DYNAMIC;SAFETY ANALYSIS

7/07/0000001-00001157/ 79
CESSION NO. 00C0164901
TLE ALTERNATIVE ENERGY TECHNOLOGIES AND SYSTEMS SESSION
TE 1980
PE L
MD 9 PPS, PP. 15 THRU 24 OF TRANS. OF THE AMERICAN NUCLEAR SOCIETY, VOL. 35, FROM 1980 WINTER MEETING; WASHINGTON, D.C., NOV. 16-21, 1980
AIL AVAILABILITY - AMERICAN NUCLEAR SOCIETY PUBLICATIONS, 555 N. KENSINGTON AVE., LA GRANGE PARK, ILL. 60525
TEGORY 010000;230000
ITION 0126
UNTRY A
STRACT THIS SESSION INCLUDED NINE ABSTRACTS OF PAPERS DISCUSSING: THE INDUSTRY ON CANVEY ISLAND AND ITS POTENTIAL HAZARDS, ASSESSMENT OF ADVANCED COAL-BASED ENERGY CONVERSION TECHNOLOGIES POINTS WAY TOWARD RATIONAL ENVIRONMENTAL R&D PROGRAM, ENERGY IS GOOD FOR YOUR HEALTH: A PLEA FOR COMPLETENESS IN HEALTH IMPACT ASSESSMENTS, ASSESSMENT OF ACTUAL AND PERCEIVED RISKS OF ENERGY DEVELOPMENT, RISKS IN ENERGY GENERATION AND SOCIETY'S VALUATION OF LIFE SAVING, ALTERNATIVE ENERGY SYSTEMS RISK ASSESSMENT, ENERGY PLANNING IN DEVELOPING AND INDUSTRIALIZING COUNTRIES, NATIONAL ENERGY RD&D PROGRAMS FROM AN INTERNATIONAL PERSPECTIVE, AND ALTERNATIVE ENERGY TECHNOLOGIES AND DEVELOPING COUNTRY NEEDS.
YWORDS ELECTRIC POWER, ALTERNATE;COMPARISON, FACILITIES;ENERGY SOURCE; HAZARD, RELATIVE;UNITED KINGDOM;R AND D PROGRAM;COAL;RADIATION, PUBLIC EDUCATION;RISK;INTERNATIONAL;FRANCE

7/07/0000001-00001157/ 80
CESSION NO. 00J0164883
TLE UNDERGROUND CONSTRUCTION OF NUCLEAR POWER REACTORS
THOR(S) PINTO S
RPAUTH SWISS FEDERAL INST. FOR REACTOR RESEARCH, WURENLINGEN, SWITZERLAND
TE 1980
PE C
MD 18 PPS, 2 TABS, 8 FIGS, NUCLEAR ENGINEERING & DESIGN, 61(3), PP. 441-58 (DEC. 1980)
EGORY 020000;180000;230000
ITION 0126
UNTRY Z
SHB NEDE
STRACT THIS PAPER SUMMARIZES THE MAIN FINDINGS OF A COMPREHENSIVE STUDY OF THE UNDERGROUND SITING OF NUCLEAR POWER PLANTS CARRIED OUT AT THE SWISS FEDERAL INSTITUTE FOR REACTOR RESEARCH. MAIN AIM OF THE INVESTIGATIONS MADE WAS TO IDENTIFY SUITABLE SITING VARIANTS AND TO EVALUATE THE FEASIBILITY, THE SAFETY POTENTIAL AND THE COST OF THE CONCEPT. TWO OF THE LAYOUTS DEVELOPED FOR THE MAIN SITING ALTERANTIVES - THE ROCK CAVITY ALTERNATIVE AND THE PIT SITING - ARE BRIEFLY DESCRIBED. IN THESE DESIGNS AN ACCIDENT MITIGATION SYSTEM BASED ON THE PRESSURE RELIEF CONCEPT, MEANT TO REDUCE THE CONSEQUENCES FOR THE PUBLIC AND THE ENVIRONMENT IN THE CASE OF EXTREME HYPOTHETICAL EVENTS, HAS BEEN PROPOSED AND AN EVALUATION OF ITS PERFORMANCES HAS BEEN MADE TO QUANTIFY THE ACHIEVABLE RISK REDUCTION.
WORDS SWITZERLAND;SITING;SITING, REACTOR;AGESTA (PWR);CONTAINMENT, UNDERGROUND;CONTAINMENT, PRESSURE RELIEF;ECONOMICS;RISK

/0/0000001-00001157/ 81
ESSION NO. 00J0164628
LE ESTIMATING CANCER RISKS FROM LOW DOSES OF IONIZING RADIATION
HOR(S) LAND CE
PAUTH NATIONAL CANCER INST., BETHESDA, MARYLAND
E 1980
E O
D 7 PPS, 4 TABS, 30 REFS, SCIENCE, 209(4462), PP. 1197-2003

(SEPT. 12, 1980)
EGORY 150000;190000;230000
TION 0125
NTRY A
SCIE
STRACT DISAGREEMENTS ABOUT THE SOMATIC RISKS FROM LOW DOSES OF IONIZING RADIATION STEM FROM TWO DIFFICULTIES FUNDAMENTAL TO THE LOGIC OF INFERENCE FROM OBSERVATIONAL DATA. FIRST, PRECISE DIRECT ESTIMATION OF SMALL RISKS REQUIRES IMPRACTICABLY LARGE SAMPLES. SECOND, PRECISE ESTIMATES OF LOW-DOSE RISKS BASED LARGELY ON HIGH-DOSE DATA, FOR WHICH THE SAMPLE SIZE REQUIREMENTS ARE MORE EASILY SATISFIED, MUST DEPEND HEAVILY ON ASSUMPTIONS ABOUT THE SHAPE OF THE DOSE-RESPONSE CURVE, EVEN WHEN ONLY A FEW OF THE PARAMETERS OF THE THEORETICAL FORM OF THE CURVE ARE UNKNOWN.
YWORDS LEUKEMIA;CANCER;RISK;EFFECT, SOMATIC;RADIATION EFFECT;RADIATION EXPOSURE, CHRONIC;LOW LEVEL RADIATION;RADIATION MODEL;DOSE; POPULATION EXPOSURE

//0/0000001-00001157/ 82
CESSION NO. 00EC163945
TLE THE EFFECTS OF NATURAL PHENOMENA ON THE EXXON NUCLEAR COMPANY MIXED OXIDE FABRICATION PLANT AT RICHLAND, WASHINGTON
RPAUTH U.S. NUCLEAR REGULATORY COMMISSION
TE 1980
PE H
RO NUREG-0722 +. 42 PPS. 12 TABS. 15 FIGS. 18 REFS. SEPT. 1980
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 130000;010000;230000;020000;150000
ITION 0124
RP CODE NRC
UNTRY A
STRACT AN ANALYSIS OF THE EFFECTS OF NATURAL PHENOMENA ON THE EXXON NUCLEAR COMPANY MIXED OXIDE FABRICATION PLANT AT RICHLAND, WASHINGTON HAS BEEN PREPARED BY THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS. THE ANALYSIS IS IN SUPPORT OF THE SPECIAL NUCLEAR MATERIALS LICENSE HELD BY THE SUBJECT COMPANY. IT ADDRESSES THE PROBABLE EFFECTS OF DAMAGE TO THE EXXON NUCLEAR COMPANY MIXED OXIDE FABRICATION PLANT BY SEVERE WEATHER AND EARTHQUAKE AND EXPRESSES THE CONSEQUENCE OF DAMAGE AS DOSE TO SEVERAL HUMAN RECEPTORS. THE DOSES THAT RESULT FROM FACILITY DAMAGE ARE MULTIPLIED BY THE OCCURRENCE RATE FOR THE INITIATING EVENT YIELDING THE YEARLY RISK.
YWORDS FUEL REPROCESSING;LICENSING PROCESS;ENVIRONMENT;HOT CELL;SITING; DOSE;SEISMOLOGY;WIND;FUEL RECYCLE;PLUTONIUM;RISK;BENEFIT VS RISK;MIXED OXIDE;FABRICATION FACILITY;REPORT, ENVIRONMENTAL

//0/0000001-00001157/ 83
CESSION NO. 00X0163783
LE REPORT OF THE ZION/INDIAN POINT STUDY: VOLUME 1
HOR(S) NORFIN WB
RPAUTH SANDIA NATIONAL LABS., ALBUQUERQUE, N.M.
TE 1980
PE N
RO NUREG/CR-1410 + SAND80-0617/1 +. APPROX. 400 PPS. FIGS. REFS. AUG. 1980
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 170000;180000;120000;080000;230000
TION 0124
P CODE AOA
NTRY A
TRACT THIS REPORT CONTAINS DETAILED RESULTS OF A STUDY FOR THE IDENTIFICATION OF REACTOR CORE-MELT ACCIDENT MITIGATION MEASURES AT THE ZION AND INDIAN POINT PLANTS. MITIGATION STRATEGIES HAVE BEEN IDENTIFIED THAT SHOW PROMISE OF PROVIDING LARGE REDUCTION IN CONSEQUENCES FOR SPECIFIC ACCIDENT SEQUENCES. HOWEVER, WITHOUT AN OVERALL RISK ANALYSIS, IT IS NOT CLEAR TO WHAT EXTENT A GIVEN MITIGATION SCHEME REDUCES OVERALL RISK. THE STUDY EVALUATED FILTERED-VENTED CONTAINMENT SYSTEMS, STEAM EXPLOSIONS, HYDROGEN BURNING, HYDROGEN CONTROL MEASURES, MELT/CONCRETE AND MELT/MGO INTERACTIONS, AND MELTDOWN PHENOMENOLOGY.

ORDS HJCK;NRC-7;ZION 1 (PWR);ZION 2 (PWR);INDIAN POINT 1 (PWR); INDIAN POINT 2 (PWR);INDIAN POINT 3 (PWR);REACTOR, PWR;CORE MELTDOWN;RISK;POPULATION EXPOSURE;EXPLOSION;HYDROGEN; CONTAINMENT FILTERING SYSTEM;RADIOACTIVITY RELEASE;ACCIDENT, LOSS OF COOLANT

/0/0000001-00001157/ 84
ESSION NO. 0000163319
LE INTEGRATION OF CAD/CAM SYSTEMS FOR PROTECTION OF STRUCTURAL COMPONENTS
HOR(S) SANDERSON RJ
PAUTH GRUMMAN AEROSPACE CORP., BETHPAGE, N.Y.
E 1980
E L
 10 PPS, PP. 203-12, BOOK PUBLISHED BY PLENUM PRESS, N.Y., 1980 (ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG. 21-26, 1977)
EGORY 110000;090000;230000
TION 0123
NTRY A
STRACT WITH THE ADVENT OF COMPUTER GRAPHICS, A TECHNOLOGY IS AFFORDED WHEREBY PRODUCT DESIGN IS TRULY DESCRIBED BY A SINGLE SOURCE. THIS SOURCE, RESIDING AS A MATHEMATICAL MODEL IN A COMPUTER, IS ACCESSED TO DISCRETELY DESCRIBE ALL MATING PARTS AND TO GENERATE MACHINE PARTS TO EITHER FABRICATE THE PARTS OR TO FABRICATE THE TOOLS REQUIRED TO MAKE THE PART. THE FAMILY OF TOOLS IS REDUCED OR ENTIRELY ELIMINATED IN THE PROCESS. THE NET RESULT IS AN IMPROVEMENT IN COMPONENT PARTS QUALITY AND IN AN IMPROVED ASSEMBLY WHICH IS MORE LIKELY TO PERFORM ITS FUNCTION. THIS CHAPTER DESCRIBES THIS TECHNOLOGY AND SHOWS HOW IT IS BEING IMPLEMENTED TO IMPROVE PRODUCT QUALITY. (EWH)
WORDS FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;EQUIPMENT DESIGN;ANALYTICAL MODEL;DESIGN STUDY;FABRICATION;EQUIPMENT

/0/0000001-00001157/ 85
ESSION NO. 0000163318
LE MICROCIRCUIT RELIABILITY CHARACTERIZATION
HOR(S) NARESKY JJ
PAUTH GRIFFISS AFB, ROME, N.Y.
E 1980
E L
 22 PPS, PP. 281-302, BOOK PUBLISHED BY PLENUM PRESS, N.Y., 1980 (ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG. 21-26, 1977)
EGORY 110000;090000;230000
TION 0123
NTRY A
TRACT SEMICONDUCTOR MICROCIRCUITS PRESENT A PARTICULARLY DIFFICULT CHALLENGE, IN TERMS OF CHARACTERIZING THEIR RELIABILITY. CONVENTIONAL RELIABILITY ANALYSIS PROCEDURES ARE INADEQUATE, AND NEW TEST METHODS AND INSTRUMENTATION MUST BE DEVELOPED. THE RELIABILITY PHYSICS APPROACH TO RELIABILITY CHARACTERIZATION IS DISCUSSED; IT IS CONCERNED WITH DEVELOPING A THOROUGH UNDERSTANDING OF THE BASIC FAILURE MECHANISMS IN THESE DEVICES, AND HOW THEY PROCEED WITH TIME AND STRESS. THE MATHEMATICAL MODELS FOR THIS APPROACH ARE REVIEWED. ALSO DESCRIBED IS THE USE OF THE [illegible] IN DEVELOPING SCREENING TESTS TO "SCREEN OUT" A PARTICULAR FAILURE MECHANISM, AND TESTS TO ACCELERATE A PARTICULAR MECHANISM TO CAUSE DEVICE FAILURE. TYPICAL MICROCIRCUIT FAILURE MECHANISMS ARE DESCRIBED. (EWH)
WORDS FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;INSTRUMENT, COMPONENT;INSTRUMENT, CONTROL;COMPUTER CONTROL;INSTRUMENT, DIGITAL;SOLID STATE DEVICE

/0/0000001-00001157/ 86
ESSION NO. 0000163317
LE EFFECT OF WEAR ON PERFORMANCE AND RELIABILITY
HOR(S) SUH NP;SAKA N
PAUTH MASS. INST. OF TECHNOLOGY, CAMBRIDGE
 1980

L
20 PPS. PP. 243-62. BOOK PUBLISHED BY PLENUM PRESS. N.Y.. 1980
(ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY
MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR
IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG.
21-26, 1977)

TEGORY 110000;230000
ITION 0123
RP CODE MEM
UNTRY A
STRACT MANY SYSTEMS CONSISTING OF AN ASSEMBLY OF MECHANICAL PARTS IN
RELATIVE MOTION FAIL DUE TO THE DAMAGE DONE TO THE SLIDING AND
ROLLING SURFACES BY WEAR. FAILURE OF A SYSTEM MAY BE
PRECIPITATED BY JUST THE DIMENSIONAL LOSS CAUSED BY WEAR, OR
WEAR-INDUCED FATIGUE AND FRACTURE OF SUCH MACHINE ELEMENTS AS
BEARINGS, GEARS, AND SPLINES. THE SUDDEN FAILURE OF MACHINES
MAY BE PREVENTED THROUGH ON-LINE MONITORING OF THE WEAR
PROCESS, WHICH MAY BE ACCOMPLISHED BY CHECKING THE DENSITY AND
THE NATURE OF THE WEAR PARTICLES AND THE DETERIORATION OF THE
LUBRICANT. (EWH)
YWORDS FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;EFFECT;WEAR;
FAILURE, EQUIPMENT

7/07/0000001-00001157/ 67
CESSION NO. 00C0163316
TLE HIGH TEMPERATURE ENVIRONMENTAL EFFECTS ON METALS
THOR(S) GRISAFFE SJ;LOWELL CE;STEARNS CA
RPAUTH NASA LEWIS RESEARCH CENTER, OHIO
TE 1980
PE L
MO 18 PPS. PP. 225-42. BOOK PUBLISHED BY PLENUM PRESS. N.Y.. 1980
(ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY
MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR
IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG.
21-26, 1977)
TEGORY 110000;230000
ITION 0123
RP CODE NSA
UNTRY A
STRACT THIS CHAPTER IS AN OVERVIEW OF PRESENT UNDERSTANDING AND
ABILITY TO PREDICT HIGH TEMPERATURE ENVIRONMENTAL ATTACK OF
METALS. THE GAS TURBINE ENGINE IS USED AS AN EXAMPLE BUT MOST
OF THE TECHNOLOGY APPLIES EQUALLY TO OTHER SYSTEMS. (EWH)
YWORDS FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;EFFECT;HIGH
TEMPERATURE;ENVIRONMENT;METAL;FAILURE;TURBINE

7/07/0000001-00001157/ 38
CESSION NO. 00C0163315
TLE ENVIRONMENTALLY ASSISTED FAILURES IN ORDANCE COMPONENTS
THOR(S) THORNTON PA;COLANGELO VJ
RPAUTH WATERVLIET ARSENAL, N.Y.
TE 1980
PE L
MO 22 PPS. PP. 203-24. BOOK PUBLISHED BY PLENUM PRESS. N.Y.. 1980
(ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY
MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR
IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG.
21-26, 1977)
TEGORY 110000;230000
ITION 0123
NTRY A
TRACT TO DEMONSTRATE THE INTERACTION BETWEEN FAILURE ANALYSES AND
DETRIMENTAL ENVIRONMENTS, BOTH IN MANUFACTURING AND IN SERVICE,
THIS CHAPTER HAS BEEN RESTRICTED TO A FEW EXAMPLES INVOLVING
ENVIRONMENTALLY ASSOCIATED FAILURES. CASE HISTORIES ARE
PRESENTED DEALING WITH THE FOLLOWING TYPE FAILURES IN STEEL:
LIQUID METAL EMBRITTLEMENT, HYDROGEN EMBRITTLEMENT, PITTING
CORROSION AND STRESS CORROSION, WHICH HAVE OCCURRED IN WEAPON
COMPONENTS. THESE CASES ARE REVIEWED IN SOME DETAIL TO CONVEY
THE TECHNIQUES UTILIZED IN ARRIVING AT THE REASON(S) FOR
FAILURE. (EWH)
WORDS FAILURE MODE ANALYSIS;RISK;EFFECT;ENVIRONMENT;FAILURE,
COMPONENT;CORROSION;STRESS CORROSION;EMBRITTLEMENT;RELIABILITY,
COMPONENT

001-0000115// 89
NO. 00C0163314
A NOTE ON FATIGUE SCATTER AND LIFE PREDICTIONS
WEISS V;KUO A
SYRACUSE UNIV., N.Y.
1980
L
8 PPS. PP. 195-202. BOOK PUBLISHED BY PLENUM PRESS, N.Y., 1980
(ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY
MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR
IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG.
21-26, 1977)
110000;220000
0123
SUR
A
ONE OF THE MAJOR PROBLEMS IN PREDICTING FATIGUE LIFE FOR
SERVICE COMPONENTS FROM LABORATORY DATA IS THAT THE SCATTER OF
LABORATORY DATA OFTEN DIFFERS FROM THAT OF ACTUAL SERVICE
COMPONENTS. IN THIS NOTE A METHOD TO ESTIMATE FATIGUE LIVES
FROM CRACK PROPAGATION MODELS IS PRESENTED. IT WILL BE SHOWN
THAT THE RESULT IS A SCATTER AND LIFE PREDICTION WHICH WELL
REPRESENTS THOSE DESIRED FOR TYPICAL MANUFACTURED PARTS. (EWH)
FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;FATIGUE;
FAILURE, FATIGUE;CRACK;ANALYTICAL MODEL;FORECAST

00001-0000115// 90
N NO. 00C0163313
REVIEW OF CONTEMPORARY APPROACHES TO FATIGUE DAMAGE ANALYSIS
SOCIE DF;MORROW J
UNIV. OF ILL., URBANA
1980
L
55 PPS. PP. 141-94. BOOK PUBLISHED BY PLENUM PRESS, N.Y., 1980
(ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY
MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR
IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG.
21-26, 1977)
110000;230000
0123
UIL
A
A REVIEW OF CURRENT METHODS FOR FATIGUE DAMAGE ANALYSIS
EMPLOYING SMOOTH SPECIMEN MATERIALS DATA FOR PREDICTING THE
SERVICE LIFE OF COMPONENTS AND STRUCTURES SUBJECTED TO VARIABLE
LOADING IS PRESENTED. IT IS WRITTEN FOR THE BEGINNER IN THIS
AREA OF FATIGUE DAMAGE ANALYSIS RATHER THAN FOR EXPERIENCED
PRACTITIONERS. SPECIAL EMPHASIS IS PLACED ON THE DETAILED
ELEMENTS OF THE ANALYSIS, AS WELL AS THE OVERALL PATTERN FOR
SYNTHESIZING THESE ELEMENTS INTO A WORKING COMPUTER-BASED
PROGRAM. (EWH)
FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;REVIEW;
FAILURE, FATIGUE;ANALYTICAL TECHNIQUE;COMPONENTS;STRUCTURE;
STRUCTURAL ANALYSIS, DYNAMIC

0000001-0000115// 91
ION NO. 00C0163312
FRACTURE MECHANICS APPLICATIONS FOR SHORT FATIGUE CRACKS
EL HADDAD MH;TOPPER TH
UNIV. OF WATERLOO, ONTARIO, CANADA
1980
L
20 PPS. PP. 121-40. BOOK PUBLISHED BY PLENUM PRESS, N.Y., 1980
(ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY
MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR
IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG.
21-26, 1977)
110000;230000
0123
A
THE AUTHORS HAVE DEVELOPED AN ELASTIC PLASTIC FRACTURE
MECHANICS SOLUTIONS WHICH APPEAR TO BE APPLICABLE TO BOTH SHORT
AND LONG CRACKS IN SMOOTH AND NOTCHED SPECIMENS. THESE
SOLUTIONS ARE EMPLOYED IN THIS CHAPTER TO PREDICT THE GROWTH OF
FATIGUE CRACKS IN SMOOTH AND AT NOTCHED SPECIMENS AND IN

ADDITION TO EXPLAIN THE PHENOMENON OF NON-PROPAGATING CRACKS. (EWH)
WORDS FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;FRACTURE TOUGHNESS;FATIGUE;FAILURE, FATIGUE;CRACK

7/0/0000001-00001157/ 92
CESSION NO. 00C0163311
TLE ENVIRONMENTALLY ASSISTED FRACTURING UNDER SUSTAINED LOADING
THOR(S) BROWN BF
RPAUTH THE AMERICAN UNIV., WASHINGTON, D.C.
TE 1980
PE L
MO 8 PPS, PP. 113-20, BOOK PUBLISHED BY PLENUM PRESS, N.Y., 1980 (ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG. 21-26, 1977)
TEGORY 110000;230000
ITION 0123
UNTRY A
STRACT THE FORM OF ENVIRONMENTALLY ASSISTED FRACTURE UNDER SUSTAINED LOAD WHICH IS OF MOST CONCERN IN CURRENT TECHNOLOGY IS STRESS CORROSION CRACKING, AND THAT WILL BE THE SUBJECT OF THIS CHAPTER. TWO OTHER MODES OF SUSTAINED-LOAD ENVIRONMENTAL FRACTURE ARE HYDROGEN EMBRITTLEMENT AND LIQUID METAL EMBRITTLEMENT. MANY OF THE PRINCIPLES INVOLVED IN SCC ALSO APPLY TO HYDROGEN AND LIQUID METAL CRACKING. INDEED, MUCH IF NOT ALL SCC IN HIGH STRENGTH STEELS IS REALLY HYDROGEN EMBRITTLEMENT, AND SOME INVESTIGATORS WOULD ALSO INCLUDE SCC CRACKING IN ALUMINUM ALLOYS, TITANIUM ALLOYS, SOME MAGNESIUM ALLOYS, AND EVEN AUSTENITIC STAINLESS STEELS IN THE HYDROGEN EMBRITTLEMENT CATEGORY. (EWH)
YWORDS FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;EFFECT; ENVIRONMENT;CRACK;STRESS CORROSION;FRACTURE TOUGHNESS;HYDROGEN; METAL;STEEL;EMBRITTLEMENT

7/0/0000001-00001157/ 93
CESSION NO. 00C0163310
TLE DUCTILE FRACTURE ANALYSIS AND SAFETY OF NUCLEAR PRESSURE VESSELS
THOR(S) LOSS FJ
RPAUTH NAVAL RESEARCH LAB., WASHINGTON, D.C.
TE 1980
PE L
MO 19 PPS, PP. 93-111, BOOK PUBLISHED BY PLENUM PRESS, N.Y., 1980 (ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG. 21-26, 1977)
EGORY 110000;230000
TION 0123
P CODE NRL
NTRY A
TRACT THIS CHAPTER REVIEWS CURRENT FRACTURE ANALYSIS METHODOLOGY FOR NUCLEAR VESSELS FROM THE VIEWPOINT OF LINEAR ELASTIC FRACTURE MECHANICS AND CODE PROCEDURES WHICH ARE APPLIED TO ASSURE STRUCTURAL INTEGRITY. THE CONTINUING NECESSITY TO QUANTIFY THE MARGIN OF SAFETY AGAINST FRACTURE AND TO QUALIFY EXISTING CONSERVATISMS IN THE OPERATION OF THESE CRITICAL STRUCTURES HAS SPURRED NEW RESEARCH RELATING TO THE CHARACTERIZATION OF ELASTIC-PLASTIC FRACTURE. DEVELOPMENTS IN THIS AREA ARE SUMMARIZED WITH EMPHASIS ON THE J-INTEGRAL APPROACH. ELASTIC-PLASTIC FRACTURE MECHANICS IS PLACED IN PERSPECTIVE WITH RESPECT TO THE LIKELY BENEFITS AND IMPLICATIONS OF CURRENT RESEARCH RELATING TO LIGHT WATER REACTOR SYSTEMS. (EWH)
WORDS FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;FRACTURE TOUGHNESS;PRESSURE VESSELS;DUCTILITY;ELASTICITY

/0/0000001-00001157/ 94
SSION NO. 00C0163309
E X-RAY DIFFRACTION TECHNIQUES IN ANALYSIS AND PREDICTION OF FAILURE
HOR(S) HERGLOTZ HK
PAUTH E.I. DU PONT DE NEMOURS & CO., WILMINGTON, DELAWARE

1980
L
21 PPS. PP. 53-73. BOOK PUBLISHED BY PLENUM PRESS. N.Y.. 1980 (ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y.. AUG. 21-26, 1977)
TEGORY   110000;090000;230000
ITION   0123
UNTRY   A
STRACT   THE UNDERLYING CONCEPTS OF XRD TECHNIQUES, THE COMMONLY USED INSTRUMENTATION, APPLICATIONS, LIMITATIONS, AND PITFALLS ARE REVIEWED AND ILLUSTRATED BY A FEW REPRESENTATIVE EXAMPLES. (EWH)
YWORDS   FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;ANALYTICAL TECHNIQUE;X-RAY;INSTRUMENT, OPTICAL;INSTRUMENT, SURVEILLANCE

7/07/0000001-0000115//   95
CESSION NO.   00C0163308
TLE   NDT - AN AID TO FAILURE ANALYSIS
THOR(S)   HATCH HP
RPAUTH   ARMY MATERIALS & MECHANICS RESEARCH CENTER, WATERTOWN, MASS.
TE   1980
PE   L
SO   10 PPS. PP. 43-52. BOOK PUBLISHED BY PLENUM PRESS. N.Y.. 1980 (ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y.. AUG. 21-26, 1977)
TEGORY   110000;230000
ITION   0123
UNTRY   A
STRACT   IMPROVED PERFORMANCE AND RELIABILITY ARE DEPENDENT UPON THE MATERIAL QUALITY CHARACTERISTICS OF INDIVIDUAL COMPONENTS, AND NONDESTRUCTIVE TESTING TECHNIQUES AID IN PREDICTING PREMATURE FAILURE BY THE DETECTION OF CRITICAL SIZE DEFECTS OR BY THE DETECTION OF MATERIAL PROPERTY GRADIENTS WHICH CAN BE EQUALLY DETRIMENTAL IN BRITTLE MATERIALS. HOWEVER, QUANTITATIVE NDT RESULTS ARE DEPENDENT UPON A NUMBER OF VARIABLES. TO ILLUSTRATE THE EFFECTIVENESS OF NDT TO ASSIST WITH THE ANALYSIS OF SUSPECT MATERIAL, RESULTS OF METALLURGICAL AND NDT ANALYSES OF TWO TRANSMISSION GEARS ARE PRESENTED. (EWH)
YWORDS   FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;MATERIAL; COMPONENTS;METALLURGY;TEST, NONDESTRUCTIVE

7/07/0000001-0000115//   96
CESSION NO.   00C0163307
TLE   RISK AND FAILURE ANALYSIS FOR IMPROVED PERFORMANCE AND RELIABILITY
THOR(S)   DOLAN TJ
RPAUTH   UNIV. OF ILL., URBANA
TE   1980
PE   L
SO   42 PPS. PP. 1-42. BOOK PUBLISHED BY PLENUM PRESS. N.Y.. 1980 (ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y.. AUG. 21-26, 1977)
EGORY   110000;230000
ITION   0123
P CODE   UIL
NTRY   A
TRACT   THE CARE AND PHILOSOPHY EMPLOYED IN MATERIAL SELECTION, DESIGN ANALYSES, FABRICATION, AND MAINTENANCE MUST BE SUFFICIENT TO LIMIT THE RISK OF FAILURE. FAILURE ANALYSIS REQUIRES CAREFUL SORTING OF A VARIETY OF INFORMATION TO DETERMINE HOW AND WHY A METAL PART FAILED, AND TO PREVENT A RECURRENCE. CONSIDERATION MUST BE GIVEN TO MAN-MACHINE INTERACTIONS TO PREVENT ACCIDENTS IN COMPLEX SYSTEMS. CONSIDERABLE LATITUDE IN USE AND MISUSE OF EQUIPMENT MUST BE FORESEEN IN ORDER TO PREDICT AND EVALUATE THE RESISTANCE TO EACH POSSIBLE MODE OF FAILURE. CAREFUL CONSIDERATION OF THE COMPLETE LIFE CYCLE IS NECESSARY FOR SELECTING OPTIMUM MATERIALS THAT WILL WITHSTAND THE MODIFICATIONS DUE TO PROCESSING AND SERVICE HISTORY, YET

PROVIDE MINIMUM RISK OF FAILURE WITH IMPROVED SAFETY AND RELIABILITY. (EWH)
WORDS FAILURE MODE ANALYSIS;RISK,RELIABILITY. COMPONENT;MATERIAL; DESIGN STUDY;FABRICATION;MAINTENANCE AND REPAIR;FAILURE;METAL; FAILURE, EQUIPMENT

//0/0000001-00001157/ 97
CESSION NO. 00C0163306
TLE NEUTRON RADIOGRAPHY UTILIZING SELECTED ENERGY INTERACTIONS
THOR(S) ANTAL JJ
RPAUTH ARMY MATERIALS & MECHANICS RESEARCH CENTER, WATERTOWN, MASS.
TE 1980
PE L
MO 14 PPS, PP. 327-40, BOOK PUBLISHED BY PLENUM PRESS, N.Y., 1980 (ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR IMPROVED PERFORMANCE & RELIABILITY; LAKE GEORGE, N.Y., AUG. 21-26, 1977)
TEGORY 110000;230000
ITION 0123
UNTRY A
STRACT THIS CHAPTER DESCRIBES THE WORK OF THE MATERIALS SCIENCES DIVISION OF AMMRC IN ATTEMPTING TO BRING NEW METHODS OF MATERIALS CHARACTERIZATION TO THE FORE SUCH AS NEUTRON RADIOGRAPHY. TWO PARTICULAR DEVELOPING TECHNIQUES REVIEWED: FISSION NEUTRON RADIOGRAPHY AND SUBTHERMAL NEUTRON RADIOGRAPHY. (EWH)
YWORDS FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT;NEUTRON; RADIOGRAPHY;TEST, NONDESTRUCTIVE;MATERIAL;TESTING

//0/0000001-00001157/ 98
CESSION NO. 00B0163305
TLE RISK AND FAILURE ANALYSIS FOR IMPROVED PERFORMANCE AND RELIABILITY
THOR(S) WEISS V;BURKE JJ
RPAUTH SYRACUSE UNIV., N.Y. ; ARMY MATERIALS & MECHANICS RESEARCH CENTER, WATERTOWN, MASS.
TE 1980
PE J
MO 355 PPS, BOOK PUBLISHED BY PLENUM PRESS, N.Y., 1980 (ISBN 0-306-40446-X) (PROCEEDINGS OF 24TH SAGAMORE ARMY MATERIALS RESEARCH CONFERENCE ON RISK & FAILURE ANALYSIS FOR IMPROVED PERFORMANCE AND RELIABILITY; LAKE GEORGE, N.Y., AUG. 21-26, 1977)
TEGORY 230000;110000
ITION 0123
RP CODE SOR
UNTRY A
STRACT THE ARMY MATERIALS AND MECHANICS RESEARCH CENTER IN COOPERATION WITH SYRACUSE UNIVERSITY HAS CONDUCTED THE SAGAMORE ARMY MATERIALS RESEARCH CONFERENCE SINCE 1954. THE MAIN PURPOSE OF THESE CONFERENCES HAS BEEN TO GATHER TOGETHER SCIENTISTS AND ENGINEERS WHO ARE UNIQUELY QUALIFIED TO EXPLORE IN DEPTH A SUBJECT OF IMPORTANCE TO THE DEPARTMENT OF DEFENSE, THE ARMY AND THE SCIENTIFIC COMMUNITY. THIS VOLUME, ADDRESSES THE AREAS OF TECHNIQUES OF FAILURE ANALYSIS, RISK AND FAILURE ANALYSIS FOR DESIGN AGAINST FRACTURE, RISK AND FAILURE ANALYSIS FOR DESIGN AGAINST FATIGUE, ELEVATED TEMPERATURE EFFECTS, ENVIRONMENTAL EFFECTS, SYSTEMS APPROACH TO PRODUCTION RELIABILITY INTEGRATION AND OUTLOOK - EMERGING NEEDS AND TECHNIQUES. (EWH)
WORDS MATERIAL;FAILURE MODE ANALYSIS;RISK;RELIABILITY, COMPONENT; REVIEW;R AND D PROGRAM;ANALYTICAL TECHNIQUE;FATIGUE;FAILURE, FATIGUE;DESIGN STUDY;EFFECT;TEMPERATURE

/0/0000001-00001157/ 99
ESSION NO. 00X0162980
LE AN APPROACH TO QUANTITATIVE SAFETY GOALS FOR NUCLEAR POWER PLANTS
PAUTH U.S. NUCLEAR REGULATORY COMMISSION
E 1980
E N
NUREG-0739 +. 151 PPS, TABS, FIGS, OCT. 1980
L AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S.

DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
GORY 010000;230000
TION 0122
CODE NRC
NTRY A
STRACT A POSSIBLE APPROACH TO QUANTITATIVE SAFETY GOALS FOR NUCLEAR POWER PLANTS IS PROVIDED BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS. THE REPORT CONTAINS THREE PARTS: 1. A REVIEW OF SEVERAL PROPOSALS FOR QUANTITATIVE RISK CRITERIA. 2. A PRELIMINARY PROPOSAL FOR A POSSIBLE APPROACH TO QUANTITATIVE SAFETY GOALS. 3. A BRIEF EVALUATION OF SEVERAL TECHNOLOGIES, INCLUDING NUCLEAR, IN TERMS OF THE PROPOSED CRITERIA. THE TRIAL APPROACH TO QUANTITATIVE SAFETY CRITERIA IS DIVIDED INTO TWO MAJOR TASKS: THE FIRST IS THE PREDOMINANTLY SOCIAL AND POLITICAL PROBLEM OF SETTING THE SAFETY CRITERIA, WHICH ARE TERMED DECISION RULES; THE SECOND IS THE TECHNICAL QUESTION OF ESTIMATING THE RISKS AND DECIDING WHETHER THE SAFETY CRITERIA HAVE BEEN MET. THE PROPOSED NUMERICAL VALUES FOR USE IN DECISION RULES ARE INTENDED TO SIMULATE FURTHER DISCUSSION AND EVALUATION LEADING TO THE FUTURE DEVELOPMENT OF SUITABLE RISK ACCEPTANCE LEVELS.
YWORDS N-POWER, SAFETY OF;INDUSTRIAL SAFETY;RISK;BENEFIT VS RISK;POWER PLANT, NUCLEAR;POWER PLANT, FOSSIL FUEL

7/07/0000001-00001157/ 100
CESSION NO. 00J0162731
TLE PUBLIC DEMANDS CANDOR ON RISK, RIGHT TO VOTE ON NUCLEAR ISSUES
TE 1980
PE D
MD 4 PPS, NUCLEAR INDUSTRY, 27(12), PP. 14-17 (DEC. 1980)
TEGORY 010000;230000
ITION 0122
UNTRY A
ABB NOIC
STRACT TO GAIN ACCEPTANCE FOR NUCLEAR POWER AND CREDIBILITY FOR THE NUCLEAR INDUSTRY WITH AN AMERICAN PUBLIC NOW MORE INCLINED TO VOTE ON NUCLEAR ISSUES THAN TO LEAVE DECISIONS WITH EXPERTS, THE INDUSTRY MUST FIRST ACKNOWLEDGE THE RISKS INVOLVED IN COMMERCIAL NUCLEAR OPERATIONS. THE PUBLIC GENERALLY LOOKS UPON THE MEDIA AS UNBIASED AND OBJECTIVE, AND THAT ONLY BY MAINTAINING GOOD CONTACTS WITH THE MEDIA, AND BY PLEDGING HONESTY AND OPENNESS TO THEM, CAN THE INDUSTRY REALLY MAKE PROGRESS IN PUBLIC ATTITUDE CHANGES.
YWORDS RISK;INDUSTRY, NUCLEAR;PUBLIC RELATIONS;RADIATION, PUBLIC EDUCATION

7/07/0000001-00001157/ 101
CESSION NO. 00J0162492
TLE REPORT TO THE PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND: TECHNICAL ASSESSMENT TASK FORCE REPORTS VOL. II: WASH 1400 - REACTOR SAFETY STUDY
THOR(S) BURNS RD
TE 1980
PE U
MD 23 PPS, 2 TABS, 3 FIGS, 38 REFS, PROGRESS IN NUCLEAR ENERGY, 6(1-3), PP. 117-40 (1980)
TEGORY 170000;010000;230000
TION 0122
NTRY U
STRACT EXAMINATION OF WASH 1400 SHOWS THAT IT IS RELEVANT TO THE STUDY OF THREE MILE ISLAND (TMI) ACCIDENT BECAUSE THE SEQUENCE OF FAILURES IN THE ACCIDENT ARE DISCUSSED IN THE REPORT, AND THE OCCURRENCE OF THE ACCIDENT IS CONSISTENT WITH WASH 1400 PREDICTIONS. WASH 1400 RESULTS, AND LESSONS THAT SHOULD HAVE BEEN LEARNED FROM THE REPORT, ARE DISCUSSED HERE.
WORDS INCIDENT;THREE MILE ISLAND 2 (PWR);REACTOR, PWR;RISK;ANALYTICAL MODEL;FAULT TREE ANALYSIS;ACCIDENT, PROBABILITY OF;OPERATING EXPERIENCE

7/07/0000001-00001157/ 102
CESSION NO. 00J0162078
LE BASIC PRINCIPLES AND RESULTS OF THE GERMAN RISK STUDY (IN GERMAN)
HOR(S) BAYER A;HEUSER FW

1980
G
6 PPS, FIGS, ATOMWIRTSCHAFT, 25(1), PP. 46-51 (JAN. 1980)
GUAGE GERMAN
EGORY 010000;230000
ITION 0121
UNTRY G
AND AWAK
STRACT IN JUNE 1976 THE FEDERAL MINISTRY FOR RESEARCH AND TECHNOLOGY HAD COMMISSIONED THE GESELLSCHAFT FUR REAKTORSICHERHEIT TO WRITE THE GERMAN RISK STUDY, THE FIRST PART OF WHICH HAS NOW BEEN COMPLETED AFTER THREE YEARS OF WORK AND HAS BEEN PUBLICIZED RECENTLY. THE GERMAN RISK STUDY IS AN ATTEMPT TO DEFINE THE SOCIETAL RISK POSED BY ACCIDENTS IN NUCLEAR POWER PLANTS UNDER CONDITIONS IN GERMANY. FOR THIS PURPOSE, THE ACCIDENT RATES AND THE RESULTANT HEALTH HAZARDS WERE DETERMINED. BY ADOPTING MOST OF THE BASIC PREMISES AND METHODS OF THE AMERICAN RASMUSSEN STUDY, THE GERMAN STUDY IS TO ALLOW A COMPARISON TO BE MADE WITH THE RESULTS OF THAT STUDY. (EWH)
YWORDS SAFETY ANALYSIS;RISK;COMPARISON;ACCIDENT ANALYSIS;GERMANY

770/0000001-00001157/ 103
CESSION NO. 00C0160760
TLE INTERNATIONAL NUCLEAR FUEL CYCLE EVALUATION REPORT (INFCE) ON WASTE MANAGEMENT AND DISPOSAL
RPAUTH INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA
TE 1980
PE L
MD STI/PUB/534 + INFCE/PC/2/7 +. 286 PPS, REPORT OF WORKING GROUP 7, JAN. 1980
AIL AVAILABILITY - UNIPUB, INC., P.O. BOX 433, NEW YORK, N.Y. 10016
TEGORY 140000;230000;220000;030000;150000
ITION 0118
RP CODE IAA
UNTRY I
STRACT THE FIRST PLENARY CONFERENCE OF THE INTERNATIONAL NUCLEAR FUEL CYCLE EVALUATION (INFCE), HELD IN VIENNA ON NOVEMBER 27-29, 1978 DECIDED THAT THE IAEA SHOULD PUBLISH FINAL REPORTS FROM THE 8 INFCE WORKING GROUPS. THUS, 8 VOLUMES WERE PUBLISHED ALONG WITH A NINTH SUMMARY VOLUME. THIS VOLUME REPORTS ON WASTE MANAGEMENT AND DISPOSAL (GROUP 7). THIS STUDY COMPARES THE MANAGEMENT AND DISPOSAL OF WASTES THAT ARISE IN THE GENERATION OF ELECTRICITY BY NUCLEAR FISSION FOR A REPRESENTATIVE SELECTION OF NUCLEAR FUEL CYCLES. ONE IMPACT ASSESSED IS THE RESIDUAL HAZARD TO HEALTH AND SAFETY. OTHER IMPACTS CONSIDERED ARE THE EFFECTS ON THE ENVIRONMENT, THE COSTS OF WASTE MANAGEMENT AND DISPOSAL, AND THE RISKS THAT FISSIONABLE MATERIALS IN THE WASTES MIGHT BE DIVERTED TO NON-PEACEFUL APPLICATIONS. FOLLOWING THE IMPACT ASSESSMENT, THE LEGAL AND INSTITUTIONAL ASPECTS OF NUCLEAR WASTE MANAGEMENT AND DISPOSAL ARE DISCUSSED, AS WELL AS THE ASPECTS OF SPECIAL INTEREST TO THE DEVELOPING COUNTRIES.
YWORDS IAEA;INFCE;FUEL CYCLE;WASTE MANAGEMENT;WASTE DISPOSAL; COMPARISON;ECONOMICS;THEFT/DIVERSION;SAFEGUARDS, NUCLEAR MATERIAL;LEGALISTICS;RISK;POPULATION EXPOSURE;ENVIRONMENT;WASTE TRANSPORTATION

770/0000001-00001157/ 104
ESSION NO. 00C0160759
LE INTERNATIONAL NUCLEAR FUEL CYCLE EVALUATION REPORT (INFCE) ON SPENT FUEL MANAGEMENT
PAUTH INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA
E 1980
E L
D STI/PUB/534 + INFCE/PC/2/6 +. 111 PPS, REPORT OF WORKING GROUP 6, JAN. 1980
IL AVAILABILITY - UNIPUB, INC., P.O. BOX 433, NEW YORK, N.Y. 10016
EGORY 120000;010000;170000;230000;030000
TION 0118
P CODE IAA
NTRY I
TRACT THE FIRST PLENARY CONFERENCE OF THE INTERNATIONAL NUCLEAR FUEL CYCLE EVALUATION (INFCE), HELD IN VIENNA ON NOVEMBER 27-29, 1978 DECIDED THAT THE IAEA SHOULD PUBLISH FINAL REPORTS FROM

THE 8 INFCE WORKING GROUPS. THUS, 8 VOLUMES WERE PUBLISHED ALONG WITH A NINTH SUMMARY VOLUME. THIS VOLUME REPORTS ON SPENT FUEL MANAGEMENT (GROUP 6). THE GROUP'S STUDY FOCUSED ON: (A) STORAGE STRATEGIES AND COSTS: FOR LIGHT WATER REACTORS (LWR), FOR HEAVY WATER REACTORS (HWR), FOR GAS-COOLED REACTORS (GCR), FOR FAST BREEDER REACTORS (FBR); (B) SHORT-TERM/INTERMEDIATE STORAGE: ASSESSMENT OF CURRENT STORAGE CAPABILITIES, WAYS OF INCREASING SPENT FUEL STORAGE CAPACITY, SITING AND TRANSPORTATION PROBLEMS, MORE EFFICIENT UTILIZATION OF EXISTING SPENT FUEL STORAGE CAPACITY, INSTITUTIONAL, ENVIRONMENTAL, SAFEGUARDS AND SAFETY ASPECTS INCLUDING FUEL INTEGRITY PROBLEMS AND ASSOCIATED RISKS, COSTS, LEGAL MATTERS; (C) SPECIAL NEEDS OF DEVELOPING COUNTRIES.

YWORDS IAEA;INFCE;FUEL CYCLE;SPENT FUEL;FUEL MANAGEMENT;ADMINISTRATIVE CONTROL;REACTOR, LWR;REACTOR, PWR;SITING;SPENT FUEL STORAGE, AFR;TRANSPORTATION AND HANDLING;SAFEGUARDS, NUCLEAR MATERIAL; RISK;REACTOR, BWR;REACTOR, GCR;REACTOR, FAST;REACTOR, BREEDER; REACTOR, LMFBR;FUEL STORAGE;SPENT FUEL POOL;SYSTEM CAPACITY

7/07/0000001-00001157/ 105

CESSION NO. 00C0160012
TLE MELT/CONCRETE INTERACTIONS: THE SANDIA EXPERIMENTAL PROGRAM, MODEL DEVELOPMENT, AND CODE COMPARISON TEST
THOR(S) POWERS DA;MUIR JF
RPAUTH SANDIA LABS., ALBUQUERQUE, N.M.
TE 1979
PE L
MO SAND-79-1916C + CONF-791118-3 +. 50 PPS. FROM 7TH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING; GAITHERSBURG, MARYLAND, NOV. 5-9, 1979
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 110000;230000
ITION 0118
RP CODE AUA
UNTRY A
STRACT THE PURPOSE OF THESE STUDIES WAS TO DEVELOP AN UNDERSTANDING OF THE INTERACTIONS SUITABLE FOR RISK ASSESSMENT. IN THIS PAPER, RESULTS OF THE EXPERIMENTAL PROGRAM ARE SUMMARIZED AND A COMPUTER MODEL OF MELT/CONCRETE INTERACTIONS IS DESCRIBED. A MELT/CONCRETE INTERACTION TEST THAT WILL ALLOW THIS AND OTHER MODELS OF THE INTERACTION TO BE COMPARED IS ALSO DESCRIBED. THE CODE COMPARISON EXERCISE, USING THIS SANDIA TEST AND A SIMILAR TEST AT THE KERNFORSCHUNGSZENTRUM KARLSRUHE, WAS SUGGESTED TO PROVIDE A BASIS FOR EVALUATING THE VARIOUS CODES WITH RESPECT TO HOW WELL THEY MODEL THE IMPORTANT INTERACTION PHENOMENA AND PREDICT CRITICAL EVENTS. (FAH)
YWORDS CORE MELTDOWN;RISK;SAFETY EVALUATION;EFFECT;CONCRETE;ANALYTICAL MODEL;COMPUTER PROGRAM;ACCIDENT, LOSS OF COOLANT;COMPARISON, THEORY AND EXPERIENCE

7/07/0000001-00001157/ 106

CESSION NO. 00X0160018
TLE ALLOCATION OF NRC INSPECTION EFFORT TO RISK-RELATED ACTIVITIES IN NUCLEAR POWER PLANTS
THOR(S) LYNCH CJ;BRISBIN NL;MURPHY DJ
PAUTH SANDIA LABS., ALBUQUERQUE, N.M.
E 1980
E N
O NUREG/CR-1338 + SAND80-0361 +. 133 PPS, TABS, FIGS, REFS. APRIL 1980
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 170000;230000
TION 0117
P CODE AUA
NTRY A
TRACT THE INSPECTION MODULES IN THE NRC INSPECTION PROGRAM FOR THE PREOPERATIONAL TEST, STARTUP TEST, AND OPERATIONS PHASES OF NUCLEAR POWER PLANTS WERE EXAMINED TO ASSESS WHETHER MANHOURS INVESTED IN EACH INSPECTION WERE COMMENSURATE WITH THE POTENTIAL OF THESE INSPECTIONS FOR DETECTING CONDITIONS WHICH WOULD CONTRIBUTE SIGNIFICANTLY TO RISK. NO BASIS WAS FOUND IN THIS ASSESSMENT FOR FUNDAMENTAL CHANGES TO THE INSPECTION

PROGRAM. HOWEVER, TO IMPROVE PROGRAM EFFECTIVENESS, SOME MODIFICATIONS TO SPECIFIC PARTS OF THE PROGRAM APPEAR TO BE WARRANTED.
ORDS AGENCY, NRC;RISK;INSPECTION;POWER PLANT, NUCLEAR;COST, OPERATING;COST BENEFIT;CONTAINMENT, ICE CONDENSER;EMERGENCY COOLING SYSTEM;PRESSURE RELIEF;STEAM;VALVES;VENTILATION SYSTEM; QUALITY ASSURANCE

//0/0000001-0000115// 107
CESSION NO. 0000159617
TLE WHAT PRICE SAFETY. A PROBABILISTIC COST-BENEFIT EVALUATION OF EXISTING ENGINEERED SAFETY FEATURES.
THOR(S) O'DONNEL EP
RPAUTH EBASCO SERVICES INC., N.Y.
TE 1978
PE L
MO INIS-MF-4779 +. 10 PPS. FROM MEETING ON NUCLEAR POWER REACTOR SAFETY; BRUSSELS, BELGIUM, OCT. 16-19, 1978
AIL AVAILABILITY - INIS SECTION, INTERNATIONAL ATOMIC ENERGY AGENCY, P.O. BOX 590, A-1011 VIENNA, AUSTRIA
TEGORY 010000;230000;050000
ITION 0117
UNTRY L
STRACT PROVIDES A METHOD FOR PERFORMING QUANTITATIVE COST-BENEFIT EVALUATIONS FOR NUCLEAR SAFETY CONCERNS INVOLVING ACCIDENTS OF LOW PROBABILITY AND POTENTIALLY LARGE CONSEQUENCES. IT PRESENTS AN APPLICATION OF THE METHOD TO ECCS, CONTAINMENT, EMERGENCY POWER SYSTEM AND HYDROGEN RECOMBINER SYSTEM. THIS EVALUATION PROVIDES A VALUABLE ASSESSMENT OF THE RELATIVE COST EFFECTIVENESS OF THESE FEATURES IN REDUCING ACCIDENT RISK. IT ALSO PROVIDES INSIGHT INTO THE SENSITIVITY OF COST-BENEFIT CALCULATIONS TO THE MANNER IN WHICH SAFETY FEATURES ARE SEQUENTIALLY ADDED IN DESIGN. (FAH)
YWORDS UNITED STATES;COST BENEFIT;ACCIDENT;RISK;ACCIDENT, LOSS OF COOLANT;RECOMBINERS;EMERGENCY POWER, ELECTRIC;CONTAINMENT

//0/0000001-0000115// 108
CESSION NO. 0000159810
TLE ASSESSMENT OF ACCIDENT RISKS FROM GERMAN NUCLEAR PLANTS.
THOR(S) HEUSER FW
RPAUTH GESELLSCHAFT FUR REAKTORSICHERHEIT MBH, KOELN, F.R. GERMANY
TE 1979
PE L
MO INIS-MF-5720 +. 13 PPS, 8 FIGS. FROM CSNI SPECIALIST MEETING ON REGULATORY REVIEW IN THE LICENSING PROCESS; MADRID, SPAIN, NOV. 7-9, 1979
NGUAGE GERMAN
AIL AVAILABILITY - INIS SECTION, INTERNATIONAL ATOMIC ENERGY AGENCY, P.O. BOX 590, A-1011 VIENNA, AUSTRIA
TEGORY 010000;230000;050000
ITION 0117
P CODE CRS
UNTRY S
STRACT THE GERMAN RISK STUDY PRESENTED. THE MAIN OBJECTIVES CAN BE SUMMED UP AS FOLLOWS: (A) AN ASSESSMENT OF THE SOCIETAL RISK DUE TO ACCIDENTS IN NUCLEAR POWER PLANTS WITH REFERENCE TO GERMAN CONDITIONS; (B) TO GET EXPERIENCE IN THE FIELD OF RISK ANALYSIS AND TO PROVIDE A BASIS FOR ESTIMATION OF UNCERTAINTIES; (C) TO PROVIDE GUIDANCE FOR FUTURE ACTIVITIES IN THE GERMANY REACTOR SAFETY RESEARCH PROGRAM. FINALLY SEVERAL CONCLUSIONS REACHED BY THIS STUDY ARE DISCUSSED. (FAH)
YWORDS GERMANY;ACCIDENT;RISK;POWER PLANT, NUCLEAR;ANALYTICAL TECHNIQUE; SAFETY PROGRAM

//0/0000001-0000115// 109
CESSION NO. 0000159790
TLE DATA SPECIALIZATION FOR PLANT SPECIFIC RISK STUDIES
THOR(S) APOSTOLAKIS G;KAPLAN S;GARRICK BJ
RPAUTH UNIV. OF CALIF., LOS ANGELES ; PICKARD, LOWE & GARRICK INC., IRVINE, CALIF.
TE 1980
PE U
MO 9 PPS, 5 TABS, 7 FIGS, 12 REFS. NUCLEAR ENGINEERING & DESIGN, 56(2), PP. 321-9 (FEB. 1980)

GORY   090000;230000
ION   0117
P CODE   UAV
TRY   Z
    NEDG
TRACT   BAYES' THEOREM IS USED TO DERIVE PLANT SPECIFIC DISTRIBUTIONS FOR THE FAILURE RATES OF COMPONENTS. METHODS ARE SUGGESTED FOR THE DERIVATION OF GENERIC DISTRIBUTIONS FROM INFORMATION THAT APPEARS IN THE LITERATURE. THESE DISTRIBUTIONS ARE USED AS PRIOR DISTRIBUTIONS IN BAYES' THEOREM AND THEY ARE MODIFIED USING PLANT SPECIFIC DATA. THE POSTERIOR DISTRIBUTIONS THUS DERIVED CAN BE USED AS INPUTS IN A PLANT SPECIFIC RISK ANALYSIS. (EWH)
YWORDS   RISK;FAILURE, COMPONENT;THEORETICAL INVESTIGATION;ANALYTICAL TECHNIQUE;MATHEMATICAL TREATMENT;RELIABILITY ANALYSIS

7/07/0000001-0000115//   110
CESSION NO.   00C0159748
TLE   REPORTING REQUIREMENTS FOR FIRE PROTECTION AND RISK ASSESSMENT IN NUCLEAR POWER PLANTS
THOR(S)   YEATER ML;HOCKENBURY RW;SIDERIS AG;MARIANI LP
RPAUTH   RENSSELAER POLYTECHNIC INST. ; ENGINEERING CONSULTANTS INC. ; AMERICAN NUCLEAR INSURERS
TE   1980
PE   L
MO   9 PPS. FROM 1980 ANS/ENS TOPICAL MEETING ON THERMAL REACTOR SAFETY; KNOXVILLE, TENN., APRIL 7-11, 1980
AIL   AVAILABILITY - R.W. HOCKENBURY, RENSSELAER POLYTECHNIC INST., TROY, N.Y.
TEGORY   170000;010000;230000
ITION   0117
UNTRY   A
STRACT   DURING REVIEW AND ANALYSIS OF FIRES OCCURRING IN NUCLEAR PLANTS, IT HAS BECOME CLEAR THAT IMPROVED REPORTING AND RECORDS ARE ESSENTIAL FOR FUTURE RISK MODELLING AND PROTECTIVE DESIGN AND PLANNING. CURRENT REPORTING RESULTS ARE REVIEWED AND ADDITIONAL NEEDED FACTORS AND RECORDS ARE IDENTIFIED.
YWORDS   REACTOR, POWER;POWER PLANT, NUCLEAR;FIRE PROTECTION;INCIDENT COMPILATION;FIRE;RISK

7/07/0000001-0000115//   111
CESSION NO.   00E0158952
TLE   INCREMENTAL RISK ASSESSMENT OF PERFORMANCE OF NON-SAFETY GRADE EQUIPMENT SUBJECTED TO AN ADVERSE ENVIRONMENT (ENCLOSURE 2)
RPAUTH   ATOMIC INDUSTRIAL FORUM, INC., WASHINGTON, DC
TE   1979
PE   H
MO   26 PGS. LTR W/ATTACH. TO NRC OFFICE OF NUCLEAR REACTOR REGULATION, OCT. 19, 1979
AIL   AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)
TEGORY   120000;090000;170000;230000
ITION   0115
RP CODE   AIF
UNTRY   A
STRACT   REPORT IS A GENERIC PROBABILISTIC ANALYSIS THAT ESTIMATES THE POTENTIAL PUBLIC RISK ASSOCIATED WITH CONCERNS DESCRIBED IN IE INFORMATION NOTICE 79-22. ANALYSIS IS BASED ON A COMPARISON IF ESTIMATED PROBABILITIES OF 4 RECENTLY IDENTIFIED SEQUENCES WITH THE ESTIMATED PROBABILITIES OF POTENTIAL ACCIDENT SEQUENCES IDENTIFIED IN REACTOR SAFETY STUDY. CONCLUSION REACHED THAT INCREMENTAL IMPACT UPON PUBLIC RISK OF POTENTIAL UNRESOLVED SAFETY ISSUES APPEARS TO BE SMALL. CONCLUSION PROBABLY VALID FOR ALL PWR'S UNLESS A DEPENDENCY BETWEEN INITIATING EVENT AND FAILURE OF AUXILIARY FEEDWATER SYSTEM CAN BE IDENTIFIED.
WORDS   REACTOR, POWER;PERFORMANCE;RISK;ACCIDENT, PROBABILITY OF; AGENCY, NRC;COMPARISON;OPERATING EXPERIENCE;RELIABILITY ANALYSIS;FEEDWATER;CONTROL SYSTEM;STATISTICAL ANALYSIS

/07/0000001-0000115//   112
ESSION NO.   00X0158851
LE   CANVEY: THE REACTION OF A COMMUNITY EXPOSED TO SUBSTANTIAL RISKS

R(S) CAVE L
AUTH UNIV. OF CALIF., LOS ANGELES
1980
N
UCLA-ENG-7967 + ALO-80 +. 9 PPS, 1 REF, OCT. 1979
IL AVAILABILITY - THE UNIV. OF CALIF., SCHOOL OF ENGINEERING & APPLIED SCIENCE, ENERGY & KINETICS DEPT., REPORTS GROUP, LOS ANGELES, CALIF.
TEGORY 230000
ITION 0114
RP CODE UAV
UNTRY A
STRACT THE REACTION OF THE CANVEY ISLAND POPULATION PROVIDES CONSIDERABLE SUPPORT FOR THE VIEW THAT "COMPARABILITY OF RISK" IS A CONCEPT THAT THE PUBLIC, IN A REAL SITUATION, CAN READILY COMPREHEND AND, IN THE CASE OF THE UK AT LEAST, ARE WILLING TO ACCEPT AS A REASONABLE BASIS FOR DECISIONS AFFECTING THEIR SAFETY, EVEN IF THEY DO NOT BENEFIT DIRECTLY FROM THE RISKS IN QUESTION. THE RISKS ARISE FROM OIL REFINERIES, PETROLEUM, AMMONIA, AND HYDROGEN FLUORIDE STORAGE FACILITIES, AN AMMONIUM NITRATE PLANT AND OTHER FACILITIES.
YWORDS RISK;BENEFIT VS RISK;UNITED KINGDOM;HYDROCARBON;HYDROGEN; FLUORIDE;FUEL, FOSSIL

//0/0000001-00001157/ 113
CESSION NO. 00C0158543
TLE THE OUTLOOK FOR NUCLEAR POWER
RPAUTH NATIONAL ACADEMY OF ENGINEERING
TE 1980
PE L
MO 74 PPS, PUBLISHED BY THE NATIONAL ACADEMY OF SCIENCES, NATIONAL ACADEMY OF ENGINEERING, 1980 (ISBN 0-309-03039-0) (FROM TECHNICAL SESSION OF THE ANNUAL MEETING; WASHINGTON, D.C., NOV. 1, 1979)
AIL AVAILABILITY - PRINTING & PUBLISHING OFFICE, NATIONAL ACADEMY OF SCIENCES, 2101 CONSTITUTION AVE., N.W., WASHINGTON, D.C. 20418
TEGORY 010000;230000;140000
ITION 0114
UNTRY A
STRACT PRESENTS PAPERS THAT WERE GIVEN AT THE TECHNICAL SESSION OF THE ANNUAL MEETING OF THE NATIONAL ACADEMY OF ENGINEERING (NOV. 1, 1979). THE TOPICS INCLUDED: (1) NEED FOR NUCLEAR POWER WORLDWIDE: WORLD REGIONAL ENERGY MODELING; (2) RISK AND DEMOCRACY; (3) NUCLEAR POWER RELIABILITY AND SAFETY IN COMPARISON TO OTHER MAJOR TECHNOLOGICAL SYSTEMS; (4) THE ELECTRIC INDUSTRY'S RESPONSE TO CURRENT EVENTS; (5) NUCLEAR WASTE MANAGEMENT; AND (6) FUTURE NUCLEAR SYSTEMS TECHNOLOGY.
YWORDS N-POWER, SAFETY OF;RISK;RELIABILITY ANALYSIS;INDUSTRY, UTILITY; WASTE MANAGEMENT;ENERGY POLICY;COMPARISON;FUEL, FOSSIL;SYSTEM ANALYSIS

//0/0000001-00001157/ 114
CESSION NO. 00J0158218
TLE ESTIMATING THE COSTS OF HYPOTHETICAL REACTOR ACCIDENTS
THOR(S) HSIEH K;SPINRAD BI
RPAUTH OREGON STATE UNIV.
E 1979
PE O
MO 8 PPS, 4 TABS, 2 FIGS, ANNALS OF NUCLEAR ENERGY, 6(7-8), PP. 445-52 (1979)
EGORY 010000;230000
TION 0113
P CODE 050
NTRY A
BB ANSE
TRACT RADIOACTIVE RELEASE FROM A REACTOR ACCIDENT IS CONSIDERED THE MAJOR RISK IN THE OPERATION OF A REACTOR. THE CONSEQUENCE OF ACCIDENT RELEASE INCLUDES VARIOUS HEALTH EFFECTS AND PROPERTY DAMAGE. IN ORDER TO EVALUATE THE ECONOMIC RISK OF REACTOR ACCIDENTS TO THE SOCIETY, THE PROBABILITY VS CONSEQUENCE CURVES IN THE REACTOR SAFETY STUDY (U.S. NRC, 1975) WERE CONVERTED TO RISK VS CONSEQUENCE CURVES FOR THE VARIOUS HEALTH EFFECTS AND PROPERTY DAMAGE (HSIEH, 1978). THIS ARTICLE WILL DEVELOP THE

METHODOLOGY TO CONVERT THE ABOVE RISKS TO PRESENT-WORTH DOLLAR VALUES, SO THAT THE RISKS TO SOCIETY OF VARIOUS CONSEQUENCES OF REACTOR ACCIDENTS CAN BE EASILY COMPARED.

WORDS REACTOR;ACCIDENT, CONSEQUENCES;ECONOMIC STUDY;RADIOACTIVITY RELEASE;RISK;CANCER;EFFECT, GENETIC;DAMAGE;STATISTICAL ANALYSIS

//0/0000001-00001157/ 115

CESSION NO. 0000158199

TLE HOW SAFE IS "TOO" SAFE?

THOR(S) BLACK SCHNIEPAUS F

RPAUTH INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA

ATE 1980

YPE U

EMO 11 PPS, 3 TALS, 3 FIGS, 13 REFS, IAEA BULLETIN, 22(1), PP. 46-50 (FEB. 1980)

ATEGORY 010000;220000

DITION 0113

RP CODE IAA

UNTRY I

ABS IAEA

STRACT THIS PAPER SUGGESTS THAT TOTAL RISK CANNOT BE REDUCED BEYOND ANY GIVEN LIMIT. AT A CERTAIN POINT THE OCCUPATIONAL AND PUBLIC RISK OF PRODUCING SAFETY EQUIPMENT BECOMES HIGHER THAN THE REDUCTION ACHIEVED IN AN EXISTING RISK. BASED ON DATA FROM THE FEDERAL REPUBLIC OF GERMANY, IT HAS BEEN ESTIMATED THAT 1 EQUIVALENT DEATH OR 6000 EQUIVALENT LOST MAN-DAYS ARE CAUSED DURING THE CONSTRUCTION AND INSTALLATION OF SAFETY EQUIPMENT COSTING ABOUT $33 MILLION. THUS, EXPENDITURES ON SAFETY AT MARGINAL COSTS OF RISK REDUCTION HIGHER THAN $33 MILLION PER EQUIVALENT LIFE SAVED WOULD ACTUALLY LEAD TO AN INCREASE IN RISK. THIS EXPENDITURE IMPLIES THAT 1400 MAN-YEARS OF EFFORT PER EQUIVALENT LIFE HAVE BEEN USED FOR NO NET GAIN IN SAFETY.

YWORDS IAEA;AUSTRIA;COST BENEFIT;SAFETY MARGIN;RISK;ECONOMICS

SION NO. 00C0169908
E A METHODOLOGY FOR COMPARING THE HEALTH EFFECTS OF ELECTRICITY GENERATION FROM URANIUM AND COAL FUELS
HOR(S) RHYNE WR;EL-BASSIONI AA
PAUTH H&R TECHNICAL ASSOCIATES INC., OAK RIDGE, TN ; U.S. NUCLEAR REGULATORY COMMISSION
TE 1981
PE L
MO 20 PPS. FROM WORKSHOP ON ASSESSING HEALTH IMPACTS OF ENERGY TECHNOLOGIES AT THE REGIONAL OR NATIONAL LEVEL; UPTON, NY, DEC. 7-8, 1981
AIL AVAILABILITY - W.R. RHYNE, H&R TECHNICAL ASSOCIATES INC., 977 OAK RIDGE TURNPIKE, OAK RIDGE, TN 37830
ATEGORY 230000
DITION 0135
UNTRY A
STRACT A METHODOLOGY WAS DEVELOPED FOR COMPARING THE HEALTH RISKS OF ELECTRICITY GENERATION FROM URANIUM AND COAL FUELS. THE HEALTH EFFECTS ATTRIBUTABLE TO CONSTRUCTION, OPERATION AND DECOMISSIONING OF EACH FACILITY IN THE TWO FUEL CYCLES WERE CONSIDERED. THE METHODOLOGY IS BASED ON DEFINING (1) REQUIREMENT VARIABLES FOR THE MATERIALS, ENERGY ETC., (2) EFFLUENT VARIABLES ASSOCIATED WITH REQUIREMENT VARIABLES AS WELL AS WITH FUEL CYCLE FACILITY OPERATION AND (3) HEALTH IMPACT VARIALLES FOR EFFLUENTS AND ACCIDENTS.
YWORDS PROBABILISTIC RISK ASSESSMENT;SOCIO/PHILOSOPHICAL CONSIDERATION; HAZARDS ANALYSIS;HAZARD, RELATIVE;EPRI;SAFETY ANALYSIS

8/07/0000001-00000767/ 2
CESSION NO. 00C0169907
TLE USING INPUT-OUTPUT ANALYSIS TO CALCULATE OCCUPATIONAL HEALTH AND SAFETY EFFECTS OF ENERGY TECHNOLOGIES
RPAUTH ENERGY & RESOURCE CONSULTANTS INC., BOULDER, CO
TE 1978
PE L
MO 20 PPS. FROM WORKSHOP ON ASSESSING HEALTH IMPACTS OF ENERGY TECHNOLOGIES AT THE REGIONAL OR NATIONAL LEVEL; UPTON, NY, DEC. 7-8, 1981
AIL AVAILABILITY - ENERGY & RESOURC UNSULTANTS INC., P.O. DRAWER O, BOULDER, CO 80306
TEGORY 230000
ITION 0135
UNTRY A
STRACT PRESENTS 21 VIEWGRAPHS WHICH OUTLINE THE METHODOLOGY USED IN THE ANALYSIS.
YWORDS SAFETY EVALUATION;SYSTEM ANALYSIS;ECONOMIC STUDY;MODEL, DETERMINISTIC;A CIDENT MODEL

8/07/0000001-00000767/ 3
CESSION NO. 00C0169906
TLE A COMPARISON OF INPUT-OUTPUT AND PROCESS ANALYSIS
THOR(S) MOSKOWITZ PD;ROWE MD
RPAUTH BROOKHAVEN NATIONAL LAB., UPTON, NY
TE 1981
PE L
MO 15 PPS. FROM WORKSHOP ON ASSESSING HEALTH IMPACTS OF ENERGY TECHNOLOGIES AT THE REGIONAL OR NATIONAL LEVEL, UPTON, NY, DEC. 7-8, 1981
AIL AVAILABILITY - MICHAEL D. ROWE, BIOMEDICAL & ENVIRONMENTAL ASSESSMENT DIVISION, DEPT. OF ENERGY & ENVIRONMENT, BROOKHAVEN NATIONAL LAB., UPTON, NY 11973
TEGORY 230000
ITION 0135
P CODE B2A
UNTRY A
TRACT ATTEMPTS TO COMPARE RESULTS FROM INPUT-OUTPUT AND PROCESS MODELS HAVE BEEN HAMPERED BY INHERENT DIFFERENCES IN THE MODELS AND BY INCONSISTENCIES IN ASSUMPTIONS. IN ORDER TO EXAMINE THE INTERCOMPARABILITY OF RESULTS FROM THESE DIFFERENT APPROACHES, ANALYSIS WERE PREPARED FOR A SINGLE ENERGY SYSTEM USING A CONSISTENT SET OF DATA FOR BOTH MODELS.
WORDS SYSTEM ANALYSIS;HAZARDS ANALYSIS;ECONOMIC STUDY;SAFETY PRINCIPLES AND PHILOSOPHY;ACCIDENT MODEL;MODEL, DETERMINISTIC

SSION NO. 00V0169688
E NRC DEFERS RULEMAKING ON DEVELOPMENT OF MANDATORY SYSTEM TO COLLECT OPERATIONAL DATA FROM NUCLEAR POWER PLANTS
PAUTH U.S. NUCLEAR REGULATORY COMMISSION
E 1981
E M
U NRC NEWS RELEASE 81-162 +. 1 PG. FOR WEEK ENDING OCT. 6, 1981
AIL AVAILABILITY - NRC, OFFICE OF PUBLIC AFFAIRS, WASHINGTON, D.C. 20555
TEGORY 010000;230000
ITION 0135
RP CODE NRC
UNTRY A
STRACT THE SYSTEM, PROPOSED IN JANUARY OF THIS YEAR, WOULD HAVE COMBINED THE NRC'S REPORTING SYSTEM--LICENSEE EVENT REPORTS--AND THE UTILITY INDUSTRY'S SYSTEM--THE NUCLEAR PLANT RELIABILITY DATA SYSTEM (NPRDS)--INTO A SINGLE MANDATORY REPORTING SYSTEM. PRIOR TO PROPOSING THE RULEMAKING IN JANUARY, THE NRC STAFF HAD IDENTIFIED TWO PRINCIPAL SHORTCOMINGS IN NPRDS--UTILITIES INTERPRETING THE REPORTABLE DATA DIFFERENTLY AND LOW LEVEL PARTICIPATION BY THE UTILITIES.
YWORDS AGENCY, NRC;REGULATION, NRC;POWER PLANT, NUCLEAR;RELIABILITY, SYSTEM;INDUSTRY, UTILITY;RELIABILITY, COMPONENT;DATA COLLECTION; REPORT, OPERATIONS

8/07/0000001-00000767/ 5
CESSION NO. 00X0169753
TLE CRITERIA FOR SAFETY-RELATED NUCLEAR POWER PLANT OPERATOR ACTIONS: INITIAL PRESSURIZED WATER REACTOR (PWR) SIMULATOR EXERCISES
THOR(S) BOTT TF;CROWE C;HAAS PM
RPAUTH OAK RIDGE NATIONAL LAB., TN
TE 1981
PE N
MO NUREG/CR-1908 + ORNL/NUREG/TM-434 +. 103 PPS. FIGS. REFS. SEPT. 1981
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 170000;230000;090000
ITION 0135
RP CODE F2C
UNTRY A
STRACT NUCLEAR PLANT CONTROL ROOM SIMULATOR EXERCISES FOR SEVEN PWR EVENTS WERE CONDUCTED WITH TEN CONTROL ROOM TEAMS. OPERATOR PERFORMANCE WAS RECORDED BY AN AUTOMATIC PERFORMANCE MEASUREMENT SYSTEM (PMS) AND BY SUBJECTIVE EVALUATION. RESPONSE TIMES AND ERROR PROBABILITIES WERE ESTIMATED FOR SELECTED ACTIONS. THE DATA COLLECTED WILL LATER BE COMPARED TO FIELD DATA BEING COLLECTED FOR SIMILAR EVENTS IN ORDER TO PROVIDE A BASIS FOR EXTRAPOLATION OF SIMULATOR DATA TO ACTUAL OPERATING CONDITIONS. ULTIMATELY, A BASE OF HUMAN PERFORMANCE DATA WILL BE DEVELOPED FROM SIMULATOR EXPERIMENTS WHICH CAN BE USED TO ESTABLISH CRITERIA AND STANDARDS, EVALUATE EFFECTS OF KEY PERFORMANCE-SHAPING FACTORS, AND SUPPORT SAFETY/RISK ASSESSMENT ANALYSES. (FAH)
YWORDS CONTROL PANEL/ROOM;SIMULATION;POWER PLANT, NUCLEAR;REACTOR, PWR; HUMAN FACTORS;OPERATOR ACTION;DATA COLLECTION

8/07/0000001-00000767/ 6
CESSION NO. 00B0169677
TLE IMPLICATIONS FOR REACTOR SAFETY OF THE ACCIDENT AT THREE MILE ISLAND, UNIT 2
THOR(S) OKRENT D;MOELLER DW
RPAUTH UNIV. OF CALIF., LOS ANGELES ; HARVARD UNIV., BOSTON, MA
E 1981
E J
O 45 PPS. 90 REFS. ANNUAL REVIEW OF ENERGY, VOL. 6, PP. 43-88 (1981)
EGORY 170000;230000
TION 0135
P CODE UAV;HAC
NTRY A
TRACT IT APPEARS TO BE DIFFICULT TO DEMONSTRATE WITH A HIGH DEGREE OF CONFIDENCE THAT THE FREQUENCY OF SEVERE CORE DAMAGE OR CORE

MELT FOR REACTORS IN OPERATION OR UNDER CONSTRUCTION IS LESS THAN ONE IN A THOUSAND TO ONE IN TWO THOUSAND PER YEAR. ALSO, THERE ARE SO MANY POTENTIAL PATHS TO A SEVERE CORE DAMAGE OR CORE MELT ACCIDENT THAT IT WILL BE DIFFICULT TO MAKE THE FREQUENCY OF SUCH AN ACCIDENT SIGNIFICANTLY SMALLER, WITH A HIGH DEGREE OF CONFIDENCE.

YWORDS THREE MILE ISLAND 2 (PWR);INCIDENT;REACTOR, PWR;ACCIDENT; FEEDWATER;VALVES;CONGRESSIONAL ACTIVITY;ACRS;AGENCY, NRC; TRANSIENT;ACCIDENT, LOSS OF COOLANT;CONTROL SYSTEM;RELIABILITY, SYSTEM;PROBABILITY;SOCIO/PHILOSOPHICAL CONSIDERATION

S/0/0000001-00000767/ 7

CESSION NO. 00X0169514
ILE WAMCOM, COMMON-CAUSE METHODOLOGIES USING LARGE FAULT TREES
THOR(S) PUTNEY B
RPAUTH SCIENCE APPLICATIONS INC., PALO ALTO, CA
TE 1981
PE N
MO EPRI-NP-1851 +. 87 PPS, 6 TABS, 31 FIGS, MAY 1981
AIL AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10090, PALO ALTO, CALIF. 94303
TEGORY 090000;230000
ITION 0135
RP CODE SAI
UNTRY A
STRACT DESCRIBES THE COMPUTER CODE WAMCOM, A CODE DEVELOPED TO DEAL WITH INDIVIDUAL AND COMBINED COMMON-CAUSE EVENTS AND RANDOM-FAILURE EVENTS. WAMCOM IS DESIGNED FOR THE EXPERIENCED ANALYST WORKING WITH LARGE, COMPLEX FAULT TREES. DESCRIPTIVE CAUSE SETS OF SUSCEPTIBLE SYSTEM COMPONENTS ARE IDENTIFIED, AND A SAMPLE PROBLEM, AS WELL AS A USER'S GUIDE, IS PROVIDED. (EWH)
YWORDS EPRI;FAILURE, COMMON MODE;ANALYTICAL TECHNIQUE;FAULT TREE ANALYSIS;COMPUTER PROGRAM;FAILURE MODE ANALYSIS

S/0/0000001-00000767/ 8

CESSION NO. 0020169497
ILE MIDLAND PLANT AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS
THOR(S) OLEY DC;CATE CL;GARRICK BJ
RPAUTH PICKARD, LOWE & GARRICK INC., IRVIN, CA
TE 1980
PE Q
MO PLG-0147 +. 210 PPS, OCT. 1980 (DOCKET 50-329)
AIL AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)
TEGORY 230000;180000
ITION 0135
UNTRY A
STRACT THREE ALTERNATIVE DESIGNS ARE EXAMINED: 1) THE DOUBLE CROSSOVER CASE WHICH REPRESENTS THE CURRENT MIDLAND AFW SYSTEM DESIGN AS DESCRIBED IN THE FSAR, 2) THE BASE CASE WHICH REPRESENTS THE MIDLAND AFW SYSTEM DESIGN PRIOR TO INCORPORATION OF CERTAIN MODIFICATION, AND 3) THE THREE PUMP CASE WHICH REPRESENTS A THEORETICAL COMPARISON OF THE MIDLAND 2-100% PUMP DESIGN WITH THE 2-50%, 1-100% PUMP DESIGN UTILIZED ON SEVERAL OTHER B&W PLANTS. THE RESULTS DEMONSTRATE THE LOWER UNAVAILABILITY OF THE MIDLAND AFW SYSTEM DESIGN IN COMPARISON TO THE THREE PUMP DESIGN ANALYZED.
YWORDS MIDLAND 1 (PWR);MIDLAND 2 (PWR);RELIABILITY ANALYSIS; RELIABILITY, SYSTEM;RELIABILITY, COMPONENT;AUXILIARY;FEEDWATER; AUXILIARY COOLING;PUMPS;REACTOR, PWR

S/0/0000001-00000767/ 9

CESSION NO. 00V0169487
LE NEW STUDY BOOSTS NUCLEAR SAFETY
HOR(S) SALISBURY DF
E 1981
E M
O 1 PG, THE CHRISTIAN SCIENCE MONITOR, PG. 3 (AUG. 11, 1981)
EGORY 010000;050000;180000;230000
TION 0135
NTRY A
TRACT IF MOLTEN FUEL FELL INTO WATER AT THE BOTTOM OF THE REACTOR VESSEL, A STEAM EXPLOSION THEORETICALLY COULD RESULT. IS IT

PROBABLE, EVEN POSSIBLE, THAT A STEAM EXPLOSION COULD BLOW A HOLE IN THE HEAVY, CONCRETE DOME COVERING THE REACTOR AND SO RELEASE LARGE AMOUNTS OF RADIOACTIVE MATERIAL INTO THE ENVIRONMENT? RECENT EXPERIMENTS CONDUCTED AT SANDIA NATIONAL LABORATORIES, SUPPORTED BY WORK BEING DONE IN WEST GERMANY AND ITALY, INDICATE THAT THE RISK OF SUCH AN EXPLOSION IS 10 TO 100 TIMES LESS THAN WAS ASSUMED. "GENERALLY, THIS REDUCES STEAM EXPLOSIONS FROM A MARGINAL TO AN INSIGNIFICANT SOURCE OF RISK," MAINTAINS THE RESEARCH PROGRAM MANAGER FOR THE NRC. (FAH)

YWORDS ACCIDENT;REACTOR, PWR;STEAM;EXPLOSION;CORE MELTDOWN;TESTING; AGENCY, NRC

8/0/0000001-00000767/ 10
CESSION NO. 00E0169486
ILE ANALYSIS OF DSN DATA BANK OF INCIDENTS AND EDF'S DATA BANK OF OPERATION/MAINTENANCE ANALYSIS REPORTS (OMAR) (IN FRENCH)
THOR(S) COUDRAY R;CROS G;MATTEI JM
RPAUTH CEA DIVISION D'ETUDE ET DE DEVELOPMENT DES REACTORS, FRANCE
TE 1981
E H
MU EMT/SYST/ESN/80/343 + FRRSR-299 +.
NGUAGE OTHER LANG
AIL AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
TEGORY 170000;230000
ITION 0135
UNTRY F
STRACT THE INTENT OF THIS STUDY IS TO INVESTIGATE THE POSSIBLE CONTRIBUTIONS TO SYSTEMS RELIABILITY ANALYSIS TAKING INTO ACCOUNT THE AVAILABLE EXPERIENCE AS REPORTED IN COLLECTION OF EVENTS RELATED TO SAFETY AND AVAILABILITY OF REACTOR OPERATION. THE DATA BANKS USED ARE THE CEA/DSN'S SAFETY RELATED OCCURRENCE REPORT AND THE EDF'S OPERATION/MAINTENANCE ANALYSIS REPORT (OMAR). THE SOURCES AND THE PRINTOUT OF DATA ARE DESCRIBED. IT SHOULD BE NOTED THAT THE DATA BANKS ARE COMPLEMENTARY: IN THE CASE TREATED, ONLY A THIRD OF THE TOTAL NUMBER OF EVENTS IS COMMON TO THE TWO BANKS. THE SYSTEM USED AS AN EXAMPLE FOR THE STUDY IS THE RESIDUAL HEAT REMOVAL SYSTEM (RHR) FOR FRENCH AND AMERICAN PWR PLANTS. SEVERAL SYSTEM CONFIGURATIONS ARE BRIEFLY DESCRIBED. (EWH)
YWORDS RELIABILITY ANALYSIS;RELIABILITY, SYSTEM;OPERATING EXPERIENCE; DATA COLLECTION;REACTOR, PWR;DECAY HEAT;COOLING;COOLING SYSTEM, SECONDARY;AUXILIARY COOLING;FOREIGN EXCHANGE;INCIDENT COMPILATION;FRANCE;RHR;SHUTDOWN COOLING SYSTEM

8/0/0000001-00000767/ 11
CESSION NO. 00X0169485
ILE SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3, AUXILIARY FEEDWATER SYSTEM RELIABILITY STUDY EVALUATION
THOR(S) BRADLEY GH
RPAUTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM
E 1981
PE N
MU NUREG/CR-2153 + SAND 81-1129 +. 41 PPS, 3 FIGS, 7 REFS, OCT. 1981
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 050000;120000;180000;230000
TION 0135
P CODE AUA
NTRY A
STRACT THIS REPORT PRESENTS THE RESULTS OF THE REVIEW OF THE AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS FOR THE SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3. THE ANALYSIS WAS PREPARED FOR SOUTHERN CALIFORNIA EDISON COMPANY BY COMBUSTION ENGINEERING.
WORDS AUXILIARY;WATER;ACCIDENT ANALYSIS;SAN ONOFRE 1 (PWR);SAN ONOFRE 2 (PWR);OPERATING LICENSE PROCESS;OPERATING EXPERIENCE; RELIABILITY ANALYSIS;FEEDWATER;COOLING SYSTEM, SECONDARY

/0/0000001-00000767/ 12
ESSION NO. 0000169183
LE NUCLEAR PLANT RELIABILITY DATA SYSTEM 1980 ANNUAL REPORTS OF

CUMULATIVE SYSTEM AND COMPONENT RELIABILITY
AUTH SOUTHWEST RESEARCH INST., SAN ANTONIO, TX
1981
A
NUREG/CR-2232 +. 650 PPS, TABS, FIGS, SEPT. 1981
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S.
DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 090000;120000;170000;230000
ITION 0134
RP CODE SRI
UNTRY A
STRACT THIS VOLUME IS THE SIXTH ANNUAL REPORT TO BE ISSUED FOR THE NPRDS. AS REFLECTED IN THE "STATUS OF NPRD REPORTS IN THE DATA BASE" AND IN THE "CUMULATIVE NPRD REPORTS SUBMITTED BY YEAR" PROGRESS HAS CONTINUED DURING 1980 IN REPORTING BY UTILITY PARTICIPANTS. THE DATA BASE IS, THEREFORE, REASONABLY MATURE AND CAN PROVIDE MEANINGFUL STATISTICS. THIS NPRDS DOCUMENT INCLUDES TWO TYPES OF ANNUAL REPORTS: (1) ANNUAL REPORT OF CUMULATIVE SYSTEM RELIABILITY, (2) ANNUAL REPOT OF CUMULATIVE COMPONENT RELIABILITY. BOTH ANNUAL REPORTS PROVIDE GENERIC RELIABILITY INFORMATION ON SYSTEMS AND COMPONENTS FOR THE CUMULATIVE PERIOD FROM JULY 1974 TO DECEMBER 1980. (EWH)
YWORDS RELIABILITY, SYSTEM;RELIABILITY, COMPONENT;OPERATING EXPERIENCE; EQUIPMENT;COMPONENTS;DATA COLLECTION;INFORMATION RETRIEVAL

670/0000001-00000/6// 13
CESSION NO. 0000169164
TLE TEN YEAR REVIEW 1969-1978 REPORT ON EQUIPMENT AVAILABILITY
RPAUTH NATIONAL ELECTRIC RELIABILITY COUNCIL
TE 1981
PE A
MO 80 PPS, 1981
AIL AVAILABILITY - NATIONAL ELECTRIC RELIABILITY, GENERATING AVAILABILITY DATA SYSTEM, RESEARCH PARK, TERHUNE ROAD, PRINCETON, NJ 08540
TEGORY 090000;120000;170000;230000
ITION 0134
UNTRY A
STRACT THIS IS THE TEN YEAR REVIEW FOR THE PERIOD 1969-1978 OF THE NATIONAL ELECTRIC RELIABILITY COUNCIL (NERC) GENERATING AVAILABILITY DATA SYSTEM (GADS) ON THE PERFORMANCE OF MAJOR TYPES OF ELECTRIC POWER GENERATING UNITS. THE SOURCE OF THESE DATA ARE THE ELECTRIC UTILITIES CURRENTLY PARTICIPATING IN THE GADS PROGRAM. A LISTING OF THESE UTILITIES IS PROVIDED IN APPENDIX E OF THIS REPORT. THE REPORTED STATISTICS AND INFORMATION WERE DERIVED FROM OUTAGE AND SUMMARY REPORTS OF INDIVIDUAL UNITS, AS SUBMITTED BY THE REPORTING UTILITIES. THIS INPUT DATA WAS SUBJECTED TO VALIDATION CHECKS AND FURTHER REVIEWS BY THE REPORTING UTILITIES PRIOR TO BEING ENTERED INTO THE DATA BASE; HOWEVER, THE PRIMARY RESPONSIBILITY FOR DATA ACCURACY LIES WITH THE REPORTING UTILITIES. (EWH)
YWORDS REVIEW;RELIABILITY, SYSTEM;RELIABILITY, COMPONENT;COMPONENTS; EQUIPMENT;AVAILABILITY;DATA COLLECTION;OPERATING EXPERIENCE; ELECTRIC POWER;GENERATORS;TRANSFORMERS;BREAKER;CIRCUIT CLOSERS/INTERRUPTERS;EDISON ELECTRIC INSTITUTE

670/0000001-00000/6// 14
CESSION NO. 00X0168985
TLE V.C. SUMMER NUCLEAR STATION UNIT 1 EMERGENCY FEEDWATER SYSTEM RELIABILITY STUDY EVALUATION
THOR(S) BRADLEY GH
RPAUTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM
E 1981
PE N
MO NUREG/CR-1870 + SAND80-2869 +. 60 PPS, 4 FIGS, JUNE 1981
IL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
EGORY 120000;180000;230000
TION 0134
P CODE AOA
NTRY A
TRACT SOUTH CAROLINA GAS AND ELECTRIC HAS SATISFACTORILY COMPLIED WITH THE REQUIREMENT TO PERFORM A RELIABILITY STUDY OF THEIR AFWS. THE COMPARISON OF THE REPORTED RELIABILITY OF SUMMER'S

AFWS TO THOSE OF OPERATING PLANTS SHOWS THAT SUMMER'S
RELIABILITY IS AT THE HIGH END OF THE RANGE OF AFWS RELIABILITY
FOR OPERATING REACTORS. HOWEVER, THE REVIEWER IS NOT IN
AGREEMENT WITH THIS ASSESSMENT BECAUSE OF THE 1 X 10(-5)
RELIABILITY ALLOCATION FOR THE VALVE IN THE SINGLE LINE FROM
THE CONDENSATE STORAGE TANK TO THE AFWS PUMP HEADER. A HIGHER
VALUE OF 1 X 10(-4) AS WAS USED IN NUREG-0611 IS RECOMMENDED TO
ACHIEVE A MORE VALID COMPARISON. (FAH)
SUMMER 1 (PWR);RELIABILITY ANALYSIS;AUXILIARY;FEEDWATER;VALVES;
REACTOR, PWR

00001-00000767/ 15
NO. 00X0160978
PROCEDURES GUIDE - A GUIDE TO THE PERFORMANCE OF PROBABILISTIC
RISK ASSESSMENTS FOR NUCLEAR POWER PLANTS. REVIEW DRAFT U.S.
NUCLEAR REGULATORY COMMISSION
U.S. NUCLEAR REGULATORY COMMISSION
1981
R
NUREG/CR-2300 +. APPROX. 715 PPS, FIGS, REFS, SEPT. 28, 1981
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S.
DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
230000
0135
A
THE OBJECTIVE OF THIS PRA PROCEDURES GUIDE IS TO AID IN THE
PERFORMANCE OF PROBABILISTIC RISK ASSESSMENTS FOR NUCLEAR POWER
PLANTS. TO THIS END IT DELINEATS ACCEPTABLE ANALYTICAL
TECHNIQUES; ACCEPTABLE ASSUMPTIONS AND MODELING APPROXIMATIONS;
INCLUDING THE TREATMENT OF STATISTICAL DATA; DEPENDENT
FAILURES, AND HUMAN ERRORS; METHODS FOR TREATING UNCERTAINTIES;
ACCEPTABLE STANDARDS FOR DOCUMENTATION AND QUALITY ASSURRANCE.
15 CHAPTERS, 6 APPENDICES, NUMEROUS TABLES AND FIGURES. AV
HAZARDS ANALYSIS;HAZARD, RELATIVE;SAFETY EVALUATION;PROCEDURES
AND MANUALS;ACCIDENT, PROBABILITY OF;SOCIO/PHILOSOPHICAL
CONSIDERATION;SAFETY PRINCIPLES AND PHILOSOPHY

00001-00000767/ 16
NO. 0000168824
RELIABILITY AND RISK ANALYSIS
MCCORMICK NJ
UNIV. OF WASHINGTON, SEATTLE
1981
J
436 PPS, BOOK PUBLISHED BY ACADEMIC PRESS, JULY 1981 (ISBN
0-12-482360-2)
230000
0133
NNS
A
THIS BOOK PROVIDES AN INTRODUCTION TO THE FUNDAMENTALS OF AND
PRINCIPAL RESULTS FROM, RELIABILITY AND RISK STUDIES FOR
NUCLEAR POWER APPLICATIONS. IT IS DIVIDED INTO THREE PARTS;
PART I SUMMARIZES CONCEPTS OF RELIABILITY ENGINEERING, PART II
DISCUSSES SOCIETAL RISKS FROM NUCLEAR POWER, AND PART III
COVERS OTHER RISK ASSESSMENTS. METHODS DISCUSSED INCLUDE
FAILURE DATA, FAULT TREE AND EVENT TREE ANALYSIS, PROBABILITY
CONCEPTS AND COMPUTER PROGRAMS FOR ANALYSIS.
ACCIDENT ANALYSIS;BENEFIT VS RISK;SAFETY EVALUATION;HAZARDS
ANALYSIS;ACCIDENT MODEL;SOCIO/PHILOSOPHICAL CONSIDERATION;FAULT
TREE ANALYSIS

00001-00000767/ 17
NO. 0000168475
WATERFORD STEAM ELECTRIC STATION UNIT 3 AUXILIARY FEEDWATER
SYSTEM RELIABILITY STUDY EVALUATION
BRADLEY CH
SANDIA NATIONAL LABS., ALBUQUERQUE, NM
1981
R
NUREG/CR-2219 + SAND81-1496 +. 43 PPS, 3 FIGS, SEPT. 1981
AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S.
DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
230000;180000

ON 0133
CODE AUA
TRY A
RACT PRESENTS THE RESULTS OF THE REVIEW OF THE AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS FOR THE WATERFORD STEAM ELECTRIC STATION UNIT 3. WATERFORD'S RELIABILITY IS AT THE HIGH END OF OF THE RANGE FOR OPERATING PLANTS FOR TWO OF THE THREE TRANSIENT CONDITIONS INVOLVING LOSS OF MAIN FEED WATER, AND IN THE MEDIUM RANGE FOR THE OTHER TRANSIENT CONDITION. THESE ASSESSMENTS WERE REACHED THROUGH A COMPARISON OF THE RELIABILITY OF WATERFORD'S EMERGENCY FEEDWATER SYSTEM TO THOSE OF COMBUSTION ENGINEERING DESIGNED OPERATING PLANTS.
YWORDS WATERFORD 3 (PWR);FEEDWATER;RELIABILITY, SYSTEM;REACTOR, PWR; RELIABILITY ANALYSIS

870/0000001-00000677 18
CESSION NO. 00X0162422
TLE OCCUPATIONAL SAFETY DATA AND CASUALTY RATES FOR THE URANIUM FUEL CYCLE
THOR(S) HOY HC;O'DONNEL FR
RPAUTH OAK RIDGE NATIONAL LAB., TN
TE 1981
PE R
MO ORNL-5747 +. 80 PPS, OCT. 1981
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 230000;010000;130000
ITION 0133
RP CODE F2C
UNTRY A
STRACT THIS REPORT IS INTENDED TO PROVIDE RISK ASSESSORS AND POLICY MAKERS WITH SELF-CONSISTANT, COMPARABLE OCCUPATIONAL CASUALTY INFORMATION ON THE TECHNOLOGIES THAT MAKE UP THE URANIUM FUEL CYCLE. IT CONTAINS SEVERAL CHAPTERS EACH COVERING A SPECIFIC TECHNOLOGY FOR PREPARING, PROCESSING AND/OR USING URANIUM. INCLUDED ARE URANIUM EXTRACTION (OPEN PIT AND UNDERGROUND), URANIUM MILLING, URANIUM CONVERSION, URANIUM ENRICHMENT, REACTOR FUEL FABRICATION, GENERATION OF ELECTRICAL POWER, (SEVERAL REACTOR TYPES), TRANSMISSION LINES AND THE NECESSARY TRANSPORTATION ELEMENTS. EACH CHAPTER HAS ITS RELATED TABLES AND REFERENCES. SUMMARY TABLES ARE INCLUDED.
YWORDS SAFETY EVALUATION;HAZARDS ANALYSIS;INDUSTRY, NUCLEAR;FUEL CYCLE; DATA COLLECTION;TRANSPORTATION AND HANDLING;MILLING;MINING; ENRICHMENT FACILITY;ORE CONVERSION;ELECTRIC POWER;POWER TRANSMISSION

870/0000001-00000677 19
CESSION NO. 00J0166375
TLE EXPANDING NUCLEAR ENERGY: OPTION OR NECESSITY?
THOR(S) SHAPIRO IS
TE 1981
PE O
MO 5 PPS, PROFESSIONAL SAFETY, PP. 35-39 (SEPT. 1981)
TEGORY 230000
ITION 0133
UNTRY A
STRACT A DISCUSSION OF THE PROBLEMS ASSOCIATED WITH SELECTION OF THE METHOD OF GENERATING ELECTRICAL ENERGY. THE BASIC ENERGY SOURCES THAT CAN BE USED ARE DISCUSSED AND THE IMPACTS OF THE OVERALL ENERGY TECHNOLOGY ARE COMPARED. INCREASED USE OF BOTH COAL AND NUCLEAR IS PROPOSED. THE DU-PONT OPERATION OF THE SAVANNAH RIVER REACTORS AND DISPOSAL AREAS IS DISCUSSED IN AN OVERVIEW FASHION.
YWORDS SOCIO/PHILOSOPHICAL CONSIDERATION;SAFETY PRINCIPLES AND PHILOSOPHY;COMPARISON;BENEFIT VS RISK;SAFETY PRINCIPLES AND PHILOSOPHY;ENERGY POLICY;N-POWER, SAFETY OF

870/0000001-00000677 20
CESSION NO. 00J0166242
TLE ATOM'S EVE - ENDING THE NUCLEAR AGE
THOR(S) READER M;HARDERT RA;MOULTON GL
TE 1980
PE J
MO 270 PPS, BOOK PUBLISHED BY MCGRAW-HILL BOOK CO., NY, 1980 (ISBN

0-07-051287-6)
ORY   010000;170000;180000;230000
ION   0133
TRY   A
RACT   THE TRUE ISSUE IN THE NUCLEAR DEBATE IS NOT WHETHER ISOLATED ATOMIC REACTORS SUCH AS THE DISABLED UNIT NEAR HARRISBURG CAN BE MADE SAFE, BUT RATHER THE SORT OF LIVES PEOPLE WILL BE FORCED TO LEAD WHILE THEY TRY TO SECURE THEM. AS THE VOLUME, FREQUENCY, AND LOCATIONS OF RADIOACTIVE TRANSACTIONS INCREASE GLOBALLY, PEOPLE ARE BEGINNING TO ASK HOW THEY ARE GOING TO PROTECT THEIR LIVES, LIBERTIES, AND HAPPINESS IN A WORLD FACING NUCLEAR WEAPONS PROLIFERATION AND CONTINUAL MISHANDLING OF RADIOACTIVE MATERIALS. AS THE ESSAYS IN THIS VOLUME DEMONSTRATE, MORE IS AT STAKE IN THE NUCLEAR CONTROVERSY THAN SECURING A HANDFUL OF REACTORS.
YWORDS   N-POWER FORECAST;N-POWER, SAFETY OF;INDUSTRY, NUCLEAR;OPPONENT;SOLAR;SOCIO/PHILOSOPHICAL CONSIDERATION;WEAPON, NUCLEAR;PROLIFERATION;CONSERVATION

8/0/0000001-0000076//   21
CESSION NO.   0000167995
TLE   IODINE TABLETS AS THYROID PROTECTION AFTER A REACTOR ACCIDENT: RISK-BENEFIT-CONSIDERATION (IN GERMAN)
THOR(S)   VOLF V
RPAUTH   KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
TE   1981
PE   J
MO   6 PPS, 7 FIGS, ATOMKERNENERGIE/KERNTECHNIK, 37(1), PP. 50-55 (1981)
NGUAGE   GERMAN
TEGORY   150000;020000;230000
ITION   0132
UNTRY   G
ABB   ATKE
STRACT   IT IS RECOMMENDED THAT A RISK STUDY SHOULD BE PERFORMED IN ORDER TO CHARACTERIZE THE ROLE OF THE INDIVIDUAL COMPONENTS OF THE RADIATION BURDEN TO THE THYROID. BY ADMINISTRATION OF STABLE IODINE TO AN ENDANGERED POPULATION THE RISK DUE TO RADIOIODINE COULD BE REDUCED UP TO ABOUT ONE HUNDREDTH BUT A SIDE EFFECT OF THIS MEASURE COULD BE AN INCREASED INCIDENCE OF HYPERTHYROIDISM. THIS ADDITIONAL RISK COULD, AND SHOULD, BE ELIMINATED BY THE GENERAL INTRODUCTION OF IODINATED SALT. UNTIL THEN, THERE IS GOOD JUSTIFICATION FOR GERMAN EMERGENCY LEVEL OF THYROID DOSE FOR DISTRIBUTION OF IODINE TABLETS TO BE THREE TO TEN TIMES HIGHER THAN THAT IN OTHER STATES. (EWH)
YWORDS   IODINE;THYROID;RADIATION SAFETY AND CONTROL;BENEFICIAL USE;BENEFIT VS RISK

3/0/0000001-0000076//   22
CESSION NO.   0000167976
TLE   FLOE: A PROGRAM FOR BAYESIAN ESTIMATION OF FLOOD PROBABILITIES
THOR(S)   GENTILLON CD
RPAUTH   EG&G IDAHO INC., IDAHO FALLS
TE   1981
PE   R
MO   NUREG/CR-2259 + EGG-EA-5344 +. 122 PPS, 7 TABS, 29 FIGS, SEPT. 1981
AIL   AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY   230000;020000
ITION   0132
RP CODE   EGG
UNTRY   A
STRACT   THE ANNUAL MAXIMUM DISCHARGE IS MODELED BY A LOG GAMMA DISTRIBUTION, WITH THE CAPABILITY OF INCORPORATING A TRANSFORMATION PARAMETER TO ACCOUNT FOR A REDUCTION IN MAXIMUM ANNUAL FLOW CAUSED BY DAMS OR OTHER MEASURES TO REGULATE STREAM FLOW. FROM THE MODEL, VALUES OF THE THREE PARAMETERS AT THE LOCATION DETERMINE THE "EXCEEDANCE" AT ANY SPECIFIED DISCHARGE LEVEL (I.E., THE PROBABILITY THAT THE ANNUAL MAXIMUM DISCHARGE WILL EXCEED THE SPECIFIED DISCHARGE). HYDROLOGISTS' JUDGMENT ON BOUNDS FOR EXCEEDANCE AT TWO DISCHARGE LEVELS IS THE BASIC INFORMATION USED TO FORM A PRIOR DISTRIBUTION. THE PRIOR DISTRIBUTION IS COMBINED WITH OBSERVED DATA. THEN POSTERIOR

DISTRIBUTIONS FOR EXCEEDANCE PROBABILITIES AT SELECTED DISCHARGE LEVELS ARE CALCULATED.

ORDS COMPUTER PROGRAM;PROBABILITY;FLOOD;FLOW;DISTRIBUTION;DAM; DISCHARGE;TRANSIENT;HJCK;NRC-RG;NRC-XA

/0/0000001-0000076// 23

ESSION NO. 00V0167887

TLE NRC AWARDS GRANTS TO DEVELOP METHODS FOR ANALYZING PROBABILITY OF NUCLEAR ACCIDENTS

RPAUTH U.S. NUCLEAR REGULATORY COMMISSION

TE 1981

PE 4

MO NRC NEWS RELEASE 81-85 +. 1 PG. FOR WEEK ENDING JUNE 2, 1981

AIL AVAILABILITY - NRC, OFFICE OF PUBLIC AFFAIRS, WASHINGTON, D.C. 20555

TEGORY 220000

ITION 0132

RP CODE NRC

UNTRY A

STRACT THE GRANTS, WHICH ARE FOR $238,000 TO THE INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS (IEEE), AND $228,000 TO THE AMERICAN NUCLEAR SOCIETY (ANS), CALL FOR THE SOCIETIES TO WORK TOGETHER WITH INDIVIDUALS WHO HAVE EXPERTISE IN THIS FIELD TO DRAFT A PROCEDURES GUIDE THAT A PLANT OWNER COULD FOLLOW FOR PROBABILISTIC ANALYSIS OF ACCIDENT SEQUENCES, SYSTEM FAILURES, RADIOACTIVITY RELEASES AND ACCIDENT CONSEQUENCES. THE FINAL PRODUCT OF THE PROJECT WILL BE THE PROCEDURE GUIDE, WHICH IS EXPECTED TO BE PUBLISHED FOLLOWING THE ANS CONFERENCE IN 1982.

YWORDS AGENCY, NRC;PROBABILITY;ACCIDENT, PROBABILITY OF;POWER PLANT, NUCLEAR;PROCEDURES AND MANUALS;GUIDE

8/0/0000001-0000076// 24

CESSION NO. 00E0167677

TLE MCGUIRE NUCLEAR STATION UNIT 1 AUXILIARY FEEDWATER SYSTEM RELIABILITY STUDY EVALUATION

THOR(S) BRADLEY DH

RPAUTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM

TE 1981

PE H

MO NUREG/CR-2096 + SAND81-0678 +. 50 PPS, 2 FIGS, 7 REFS, JULY 1981

AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

TEGORY 010000;230000

ITION 0131

RP CODE AOA

UNTRY A

STRACT A COMPARISON OF THE REPORTED RELIABILITY OF MCGUIRE'S AUXILIARY FEEDWATER SYSTEM (AFWS) TO THOSE OF OPERATING PLANTS SHOWS THAT MCGUIRE'S RELIABILITY IS IN THE LOW TO MEDIUM RANGE OF THE AFWS RELIABILITY FOR OPERATING PLANTS. IF THE FLOW REQUIREMENTS WERE REDUCED OR IF OPERATOR ACTION COULD BE TAKEN TO INCREASE THE QUANTITY OF AUXILIARY FEEDWATER, MCGUIRE'S RELIABILITY WOULD BE IN THE MEDIUM TO HIGH RANGE. THE REASON FOR THE LOW RELIABILITY IS THE FACT THAT THE AFWS AT MCGUIRE IS MECHANICALLY THROTTLED TO PROVIDE PROTECTION FROM A BREAK IN THE MAIN FEEDWATER LINE TO OR RUPTURE OF A STEAM GENERATOR.

YWORDS FEEDWATER;RELIABILITY, SYSTEM;MCGUIRE 1 (PWR);REACTOR, PWR; ACCIDENT, LOSS OF FLOW;OFF SITE;ACCIDENT, LOSS OF POWER; ELECTRIC POWER;STEAM GENERATOR

/0/0000001-0000076// 25

ESSION NO. 00X0167409

LE EVIDENCE OF SIGNIFICANT BIAS IN AN ELEMENTARY RANDOM NUMBER GENERATOR (IN GERMAN)

HOR(S) BRANDL V;BORGWALDT H

PAUTH KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY

E 1981

E N

O KFK-3107 + GERRSR-710 +. VP, MARCH 1981

GUAGE GERMAN

IL AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.

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GORY          230000
TION          0131
P CODE        KSR
NTRY          C
TRACT         AN ELEMENTARY PSEUDO RANDOM NUMBER GENERATOR FOR ISOTROPICALLY
              DISTRIBUTED UNIT VECTORS IN 3-DIMENSIONAL SPACE HAS BEEN TESTED
              FOR BIAS.  THIS GENERATOR USES THE IBM-SUPPLIED ROUTINE RANDU
              AND A TRANSPARENT REJECTION TECHNIQUE. THE TESTS SHOW CLEARLY
              THAT NON-RANDOMNESS IN THE PSEUDO RANDOM NUMBERS GENERATED BY
              THE PRIMARY IBM GENERATOR LEADS TO BIAS IN THE ORDER OF 1
              PERCENT IN ESTIMATES OBTAINED FROM THE SECONDARY RANDOM NUMBER
              GENERATOR.  FORTRAN LISTINGS OF 4 VARIANTS OF THE RANDOM NUMBER
              GENERATOR CALLED BY A SIMPLE TEST PROGRAM AND OUTPUT LISTINGS
              ARE INCLUDED FOR DIRECT REFERENCE.
YWORDS        STATISTICAL ANALYSIS;ERROR ANALYSIS;DECISION ANALYSIS;
              PROBABILITY;MODEL. STOCHASTIC;GERMANY;FOREIGN EXCHANGE

8/0/0000001-0000076//     26
CESSION NO.   00X0167340
TLE           RISK OF TRANSPORTING SPENT NUCLEAR FUEL BY TRUCK
THOR(S)       ELDER HK;ANDREWS WB;RHOADS RE
RPAUTH        BATTELLE PACIFIC NORTHWEST LABS., RICHLAND, WA
TE            1978
PE            N
MO            PNL-SA-6520 + CONF-780500-44 +.  10 PPS. 2 TABS. 3 FIGS. 10
              REFS. MAY 1978
AIL           AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S.
              DEPARTMENT OF COMMERCE, SPRINGFIELD, VA.  22161
TEGORY        030000;230000
ITION         0131
RP CODE       BNR
UNTRY         A
STRACT        THE RISK METHODOLOGY USED TO EVALUATE THE RISK IN SHIPPING
              SPENT FUEL WAS INITIALLY APPLIED TO THE SHIPMENT OF PLUTONIUM
              BY TRUCK.  THE METHODOLOGY INCLUDES:  (1) A DESCRIPTION OF THE
              SPENT FUEL TRANSPORT SYSTEM, (2) IDENTIFICATION OF POTENTIAL
              RELEASE SEQUENCES, (3) EVALUATION OF THE PROBABILITIES AND
              CONSEQUENCES OF THE RELEASES, AND (4) CALCULATION AND
              ASSESSMENT OF THE RISK.
YWORDS        BENEFIT VS RISK;SPENT FUEL;WASTE TRANSPORTATION;RADIOACTIVITY
              RELEASE;TRANSPORTATION AND HANDLING

8/0/0000001-0000076//     27
CESSION NO.   00X0167331
TLE           APPLICATION OF SPACE AND AVIATION TECHNOLOGY TO IMPROVE THE
              SAFETY AND RELIABILITY OF NUCLEAR POWER PLANT OPERATIONS
RPAUTH        INTERNATIONAL ENERGY ASSOCIATES LTD., WASHINGTON, D.C.
TE            1980
PE            N
MO            DOE/TIC-11143 +.  271 PPS. FIGS. APRIL 1980
AIL           AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U.S.
              DEPT. OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY        230000
ITION         0131
UNTRY         A
STRACT        THIS REPORT INVESTIGATES VARIOUS TECHNOLOGIES THAT HAVE BEEN
              DEVELOPED AND UTILIZED BY THE AEROSPACE COMMUNITY, PARTICULARLY
              NASA THAT WOULD APPEAR TO HAVE POTENTIAL FOR CONTRIBUTING TO
              SAFETY AND RELIABILITY IN THE UNITED STATES.  IT WAS DETERMINED
              THAT THERE ARE INDEED AEROSPACE TECHNOLOGIES THAT ARE GERMAINE
              TO THE IMPROVEMENT OF SAFETY AND RELIABILITY OF NUCLEAR PLANT
              OPERATIONS.  SOME TECHNOLOGIES ARE IMMEDIATELY TRANSFERABLE
              WHILE SOME WILL REQUIRE ADAPTION.  THESE ARE LISTED AND
              APPLICATIONS DESCRIBED.
YWORDS        RELIABILITY ANALYSIS;SAFETY PROGRAM;INTRINSIC SAFETY;HAZARDS
              ANALYSIS;HUMAN FACTORS

/0/0000001-0000076//     28
CESSION NO.   00X0167327
TLE           RISK ANALYSIS METHODS DEVELOPMENT - TWELFTH QUARTERLY
              REPORT-JANUARY-MARCH 1980
PAUTH         GENERAL ELECTRIC CO., SUNNYVALE, CA
E             1980
E             N
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GEFR-14023-12 +. 57 PPS. TABS. FIGS. REFS. APRIL 1980
AVAILABILITY - LIMITATIONS ON DISTRIBUTION; SEND REQUESTS TO DOE TECHNICAL INFORMATION CENTER. P.O. BOX 62. OAK RIDGE. TENN. 37830
EGORY 230000
TION 0131
RP CODE GEC
UNTRY A
STRACT THIS REPORT DESCRIBES THE ANALYSIS DONE TO DEVELOP METHODOLOGY AND DATA NECESSARY FOR A CREDIBLE BREEDER REACTOR RISK ASSESSMENT. THE RISK ALLOCATION METHOD, AS USED IN THIS REPORT AND AS APPLIED TO LBR SAFETY, IS A SYSTEMATIC PROCEDURE FOR ESTABLISHING A SAFETY RELATED SYSTEM RELIABILITY GOAL AT MINIMUM COST.
YWORDS BENEFIT VS RISK;ECONOMIC STUDY;SOCIO/PHILOSOPHICAL CONSIDERATION;SAFETY EVALUATION;SAFETY ANALYSIS REPORT. METEOROLOGY;HAZARD. RELATIVE;HAZARDS ANALYSIS

870/0000001-00000767/ 29
CESSION NO. 00X0167325
TLE RISK ANALYSIS METHODS DEVELOPMENT - QUARTERLY REPORT APRIL-JUNE 1980
RPAUTH GENERAL ELECTRIC CO.. SUNNYVALE. CA
TE 1980
PE N
HO GEFR-14023-13 +. 173 PPS. TABS. FIGS. JULY 1980
AIL AVAILABILITY - LIMITATIONS ON DISTRIBUTION; SEND REQUESTS TO DOE TECHNICAL INFORMATION CENTER. P.O. BOX 62. OAK RIDGE. TENN. 37830
TEGORY 230000
ITION 0131
RP CODE GEC
UNTRY A
STRACT THIS REPORT DESCRIBES ANALYSIS DONE TO DEVELOP METHODOLOGY AND DATA NECESSARY FOR A CREDIBLE BREEDER REACTOR RISK ASSESSMENT. RISK ALLOCATION METHOD, AS USED IN THIS REPORT AND AS APPLIED TO LBR SAFETY, IS A SYSTEMATIC PROCEDURE FOR ESTABLISHING SAFETY RELATED SYSTEM RELIABILITY GOALS AT MINIMUM COST. THIS MODEL CONSIDERS THE CURRENT OPTIONAL DESIGNS AND SELECTS THE DESIGN OPTIONS THAT SATISFY RISK CONSTRAINTS AT MINIMUM COSTS.
YWORDS BENEFIT VS RISK;ECONOMIC STUDY;SOCIO/PHILOSOPHICAL CONSIDERATION;SAFETY EVALUATION;HAZARD. RELATIVE;HAZARDS ANALYSIS;SAFETY ANALYSIS REPORT. METEOROLOGY;INTRINSIC SAFETY

870/0000001-00000767/ 30
CESSION NO. 00X0167324
TLE A METHODOLOGY AND A PRELIMINARY DATA BASE FOR EXAMING THE HEALTH RISKS OF ELICTRICITY GENERATION FROM URANIUM AND COAL FUELS
THOR(S) EL-BASSIONI AA
RPAUTH SCIENCE APPLICATIONS INC.
TE 1980
PE N
HO NUREG/CR-1539 + ORNL/SUB-7615 + SAI-OR-80-140-01 +. 505 PPS. AUG. 1980
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE. U. S. DEPARTMENT OF COMMERCE. SPRINGFIELD. VA. 22161
TEGORY 230000;190000
ITION 0131
RP CODE F2C
UNTRY A
STRACT AN ANALYTICAL MODEL WAS DEVELOPED TO ASSESS AND EXAMINE THE HEALTH EFFECTS ASSOCIATED WITH THE PRODUCTION OF ELECTRICITY FROM URANIUM AND COAL FUELS. THE MODEL IS BASED ON A SYSTEMATIC METHODOLOGY THAT IS BOTH SIMPLE AND EASY TO CHECK. AND PROVIDES DETAILS ABOUT THE VARIOUS COMPONENTS OF HEALTH RISK. AN ITERATIVE APPROACH INVOLVING ONLY A FEW STEPS IS RECOMMENDED FOR VALIDATING THE MODEL. AFTER EACH VALIDATION STEP, THE MODEL IS IMPROVED IN THE AREAS WHERE NEW INFORMATION OR INCREASED INTEREST JUSTIFIES SUCH UPGRADING. SENSITIVITY ANALYSIS IS PROPOSED AS THE BEST METHOD OF USING THE MODEL TO ITS FULL POTENTIAL. DETAILED QUANTIFICATION OF THE RISKS ASSOCIATED WITH THE TWO FUEL CYCLES IS NOT PRESENTED IN THIS REPORT. A PRELIMINARY SET OF DATA THAT IS NEEDED TO CALCULATE

THE HEALTH RISKS WAS GATHERED, NORMALIZED TO THE MODEL FACILITIES, AND PRESENTED IN A CONCISE MANNER.
ORDS FUEL CYCLE;POWER PLANT, FOSSIL FUEL;SAFETY EVALUATION;SAFETY REVIEW;SAFETY ANALYSIS;HAZARD, RELATIVE;HAZARDS ANALYSIS

/0/0000001-00000767/ 31
ESSION NO. 0020167321
TLE DIABLO CANYON NUCLEAR POWER STATION UNIT 1 AUXILIARY FEEDWATER SYSTEM RELIABILITY STUDY EVALUATION
THOR(S) BRADLEY GH
RPAUTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM
TE 1981
PE R
40 NUREG/CR-1925 + SAND81-0242 +. 53 PPS, 4 TABS, 2 FIGS, JULY 1981
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 230000;180000;120000
ITION 0131
RP CODE AOA
UNTRY A
STRACT PACIFIC GAS AND ELECTRIC ADEQUATELY DISCUSSED THE MAJOR CONTRIBUTORS TO UNRELIABILITY FOR THE THREE CASES (1) LMFW, LOSS OF MAIN FEEDWATER, (2) LMFW/LOOP, LOSS OF MAIN FEEDWATER/LOSS OF OFFSITE POWER, AND (3) LMFW/LAC, LOSS OF MAIN FEEDWATER/LOSS OF ALL AC POWER. THE MAJOR CONTRIBUTOR IN CASE 1 AND 2 IS THE FAILURE OR INCORRECT POSITIONING OF THE CONDENSATE STORAGE TANK OUTLET VALVE COMBINED WITH NO OPERATOR ACTION TO TRIP THE AUXILIARY FEEDWATER PUMPS. IT IS CONCLUDED FOR CASES 1 AND 2 THE RELIABILITY SHOULD BE IN THE MEDIUM RANGE.
YWORDS FEEDWATER;VALVES;OFF SITE;OPERATOR ACTION;ACCIDENT, LOSS OF POWER,RELIABILITY ANALYSIS;RELIABILITY, SYSTEM

8/0/0000001-00000767/ 32
CESSION NO. 00C0167266
TLE RELATIVE CONSEQUENCES OF TRANSPORTING HAZARDOUS MATERIALS
THOR(S) FOLLWOOD RR;RHYNE WR;SIMMONS JA
RPAUTH SANDIA NATIONAL LABS., ALBUQUERQUE, NM ; SCIENCE APPLICATIONS INC., OAK RIDGE, TN
TE 1980
PE L
40 SAND80-0901C + CONF-801115-27 +. 8 PPS, FROM 6TH INTERNATIONAL SYMPOSIUM ON PACKAGING & TRANSPORTATION OF RADIOACTIVE MATERIALS, NOV. 1980
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 030000;230000
ITION 0130
RP CODE AOA;SAI
UNTRY A
STRACT THE PAPER DISCUSSES METHODS UNDER STUDY AT THE TRANSPORTATION TECHNOLOGY CENTER (TTC) TO DEVELOP A PERSPECTIVE ON HOW TECHNICAL MEASURES OF HAZARD AND RISK RELATE TO PERCEPTION OF HAZARDS, HARM, AND RISKS ASSOCIATED WITH TRANSPORTING HAZARDOUS MATERIALS.
YWORDS TRANSPORTATION AND HANDLING;HAZARD, RELATIVE

8/0/0000001-00000767/ 33
CESSION NO. 00X0167240
TLE PWR REACTOR PRESSURE VESSEL FAILURE PROBABILITIES (IN ENGLISH)
THOR(S) DUFRESNE J;LANORE JM
RPAUTH CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
E 1980
PE N
40 DSN 377 + FRNSR-282 +. 30 PPS, 11 FIGS, 11 REFS, MAY 1980
AIL AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
EGORY 110000;230000
TION 0130
P CODE CEA
NTRY F
TRACT TWO METHODS HAVE BEEN DEVELOPED IN THE PAST TO EVALUATE THE

RUPTURE PROBABILITY OF A LWR VESSEL: A STATISTICAL APPROACH USING DATA FROM CONVENTIONAL PLANTS, AND A PROBABILISTIC METHOD USING THE FRACTURE MECHANICS UNDER PROBABILISTIC FORM. AT THE PRESENT TIME, ONLY CONDITIONAL PROBABILITIES OF FAILURE CAN BE DETERMINED; THIS IS MAINLY DUE TO THE LACK OF INFORMATION ON THE PROBABILITY OF OCCURRENCE OF FAULTED CONDITIONS (LOCA-STEAM BREAK-OVER PRESSURE ETC...) AND ON THE DEFECT DETECTION PROBABILITY. (FAH)

YWORDS FRANCE;PRESSURE VESSELS;FAILURE;PROBABILITY;REACTOR, LWR; FOREIGN EXCHANGE

870/0000001-0000076// 34

CESSION NO. 00E0166945
TLE DESIGN-BASIS DOCUMENT FOR TRIPS RELATED TO REACTIVITY CHANGE IN THE EBR-II PPS (IN THE OPERATE MODE)
THOR(S) CURRAN RN;DEAN EM;BOLAND JF
RPAUTH ARGONNE NATIONAL LAB., IL
TE 1980
PE H
MO ANL-76-32 +. 68 PPS, 20 TABS, 47 FIGS, 9 REFS, DEC. 1980
AIL AVAILABILITY - LIMITATIONS ON DISTRIBUTION; SEND REQUESTS TO DOE TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830
TEGORY 060000;230000
ITION 0130
RP CODE CZA
UNTRY A
STRACT REPORT ESTABLISHES DESIGN BASIS FOR EBR-II REACTIVITY-PROTECTION SUBSYSTEM OF PLANT PROTECTION SYSTEM (PPS) IN THE REACTOR OPERATE MODE OF OPERATION. REACTIVITY-INSERTION EVENTS ARE IDENTIFIED AND GROUPED USING FAULT-TREE ANALYSIS INTO ANTICIPATED, UNLIKELY, AND EXTREMELY UNLIKELY FAULTS. DYNAMIC SIMULATIONS OF REACTIVITY-INSERTION EVENTS AND PPS ARE MADE FOR WORST-CASE DESIGN-BASIS CONDITIONS TO SHOW THAT ESSENTIAL PERFORMANCE REQUIREMENTS FOR PPS ARE MET AND THAT ADEQUATE PROTECTIVE MARGINS ARE AVAILABLE. RESULTS FROM ANALYSIS IDENTIFY DESIGN-BASIS FAULTS FOR EACH PROTECTIVE TRIP FUNCTION AND ESTABLISH TIMES PERMITTED FOR COMPLETION OF PROTECTIVE ACTION.
YWORDS EBR 1 AND 2 (RE);REACTOR PROTECTION SYSTEM;ACCIDENT, DESIGN BASIS;ACCIDENT, CONTROL ROD DROPIN;FAULT TREE ANALYSIS; SENSITIVITY ANALYSIS;RELIABILITY ANALYSIS;REACTIVITY EFFECT

870/0000001-0000076// 35

CESSION NO. 00C0166926
TLE A FEASIBILITY STUDY CONCERNING THE PROBABILISTIC APPROACH IN SEISMIC ASSESSMENT-APPLICATION TO SOUTHEASTERN FRANCE (IN FRENCH)
THOR(S) GOULA X
RPAUTH CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
TE 1980
PE L
MO DSN 337 + FRRSR-285 +. 33 PPS, 16 FIGS, OCT. 1980
NGUAGE OTHER LANG
AIL AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
TEGORY 020000;230000
ITION 0130
UNTRY F
STRACT THE SEISMICITY OF THE REGION DOES NOT APPEAR TO BE RANDOMLY DISTRIBUTED, BUT RATHER CONCENTRATED IN A SMALL NUMBER OF ZONES. THE ATTENUATION OF INTENSITY WITH EPICENTRAL DISTANCE IS EVALUATED FROM ISOSEISMAL MAPS OF TWENTY, WELL DOCUMENTED EARTHQUAKES. A SPONHEUER MODIFIED MODEL IS USED FOR ADJUSTMENTS. WHENEVER POSSIBLE, AZIMUTHAL AND LATERAL VARIATIONS OF ATTENUATION COEFFICIENTS ARE CONSIDERED. THE ADJUSTMENTS MADE SHOW ROUGHLY THE FOLLOWING TENDANCIES: SHALLOW DEPTHS (0-5 KM) FOR RHONE VALLEY EVENTS; DEPTHS RANGING FROM 5 TO 10 KM FOR DURANCE VALLEY EVENTS; GREATER DEPTHS (15-30 KM) FOR SOME ALPS EVENTS; ATTENUATION COEFFICIENTS RANGING FROM 0.001 TO 0.06 KM-1. (FAH)
YWORDS FRANCE;PROBABILITY;EARTHQUAKE RECORDS;ANALYTICAL MODEL;FORECAST; FOREIGN EXCHANGE

SION NO. 00U0166602
HOW SAFE DO YOU WANT TO BE?
OR(S) MANNING R
1981
U
U 2 PPS, THE TENNESSEE CONSERVATIONIST, XLVII(4), PP. 18-19 (AUG. 1981)
EGORY 230000
ITION 0130
UNTRY A
STRACT THIS IS A SHORT ARTICLE DESCRIBING A RATIONALE OF ENERGY TECHNOLOGY SAFETY. THE ETHICAL QUESTION OF WHICH ENERGY SOURCE IS MOST LIKELY TO MEET THE REQUIREMENTS OF SOCIETY ARE DISCUSSED AND THE QUESTION REPLACING INCREASED FUEL USE WITH CONSERVATION IS POSED. NO DATA IS GIVEN.
YWORDS SOCIO/PHILOSOPHICAL CONSIDERATION;HAZARD, RELATIVE;SAFETY PRINCIPLES AND PHILOSOPHY;SAFETY EVALUATION

5/0/0000001-0000076// 37
CESSION NO. 00X0166768
TLE A SURVEY OF SAFETY LEVELS IN FEDERAL REGULATION
THOR(S) RUMER T;LAVE L
RPAUTH CARNEGIE-MELLON UNIV., PITTSBURGH, PA ; BROOKINGS INSTITUTION, WASHINGTON, DC
TE 1981
PE N
MO NUREG/CR-2226 +. 46 PPS, REFS, JUNE 1981
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 230000
ITION 0130
RP CODE AYK
UNTRY A
STRACT EACH AGENCY REGULATING HEALTH OR SAFETY MUST SET A SAFETY GOAL, IMPLICITLY OR EXPLICITLY. IN SOME CASES CONGRESS HAS GIVEN SPECIFIC GUIDANCE; MORE GENERALLY, CONGRESS HAS GIVEN IMMENSE DISCRETION TO THE AGENCIES. EIGHT FRAMEWORKS FOR REGULATING HEALTH AND SAFETY ARE DESCRIBED. IMPORTANT ISSUES IN DECIDING WHICH FRAMEWORK TO SELECT INCLUDE: (A) THE REQUIRED AMOUNT OF DATA COLLECTION, ANALYSIS AND VALUE JUDGMENTS FOR EACH, (B) WHETHER RISK CAN BE QUANTIFIED IN EACH CASE, (C) HOW EACH FRAMEWORK AFFECTS PRIORITY SETTING, (D) THE RESIDUAL LEVEL OF UNCERTAINTY AFTER ANALYSIS, (E) SAFETY GOALS, AND (F) THE GENERAL COSTS OF REGULATING.
YWORDS SAFETY ANALYSIS;SAFETY PROGRAM;SAFETY PRINCIPLES AND PHILOSOPHY; REGULATION, FEDERAL;COMPARISON

5/0/0000001-0000076// 38
CESSION NO. 00J0166374
TLE A COST-BENEFIT COMPARISON OF NUCLEAR AND NONNUCLEAR HEALTH AND SAFETY PROTECTIVE MEASURES AND REGULATIONS
THOR(S) MAURO JJ;O'DONNELL EP
RPAUTH EBASCO SERVICES INC.
TE 1979
PE J
MO 16 PPS, 8 TABS, 3 FIGS, 28 REFS, NUCLEAR SAFETY, 20(5), PP. 525-40 (SEPT.-OCT. 1979)
TEGORY 010000;230000
ITION 0129
RP CODE AGR
UNTRY A
ABS NUSA
STRACT A COST-BENEFIT METHODOLOGY FOR NUCLEAR SAFETY CONCERNS IS PRESENTED AND APPLIED TO EXISTING NUCLEAR PLANT ENGINEERED SAFETY FEATURES. COMPARISONS IN TERMS OF INVESTMENT COSTS TO ACHIEVE REDUCTIONS IN MORTALITY RATES ARE THEN MADE BETWEEN NUCLEAR PLANT SAFETY FEATURES AND THE PROTECTIVE MEASURES AND REGULATIONS ASSOCIATED WITH NONNUCLEAR RISKS, PARTICULARLY WITH COAL-FIRED POWER PLANTS. THESE COMPARISONS REVEAL A MARKED INCONSISTENCY IN THE COST EFFECTIVENESS. A SPECIFIC EXAMPLE OF REGULATORY DISPARITY REGARDING GASEOUS EFFLUENT LIMITS FOR NUCLEAR AND FOSSIL-FUEL POWER PLANTS IS PRESENTED. IT IS CONCLUDED THAT A CONSISTENT HEALTH AND SAFETY REGULATORY POLICY BASED ON UNIFORM RISK AND COST-BENEFIT CRITERIA SHOULD BE

ADOPTED AND THAT FUTURE PROPOSED NUCLEAR REGULATORY COMMISSION REGULATORY REQUIREMENTS SHOULD BE CRITICALLY EVALUATED FROM A COST-BENEFIT VIEWPOINT. (FAH)

WORDS — COST BENEFIT;POWER PLANT, NUCLEAR;POWER PLANT, FOSSIL FUEL; SAFETY PROGRAM;PERSONNEL PROTECTIVE DEVICE;COMPARISON

3/0/0000001-00000767/ 39

CESSION NO. — 00J0166268
TLE — DYNAMIC RELIABILITY MODEL FOR THE PICKERING CLADDING STRESS CORROSION CRACKING FAILURES
UTHOR(S) — INGMAN D;GUTMAN A
ORPAUTH — TECHNION-ISRAEL INST. OF TECHNOLOGY, HAIFA
ATE — 1981
YPE — U
EMO — 2 PPS, 1 FIG, 2 REFS, NUCLEAR TECHNOLOGY, 54(1), PP. 7-8 (JULY 1981)
ATEGORY — 110000;230000
DITION — 0129
OUNTRY — X
ABB — NUAT
BSTRACT — A DYNAMIC RELIABILITY MODEL IS USED TO DESCRIBE TEST DATA ON STRESS CORROSION CRACKING FAILURES OF CLADDING RINGS CUT OF THE PICKERING POST-REACTOR FUEL. THIS MODEL PROVIDES AN EXPRESSION THAT FITS THE DATA POINTS VERY WELL. THE MAIN GOAL OF THE MODEL IS TO TAKE INTO ACCOUNT THE IN-REACTOR RELIABILITY DROP PRIOR TO THE TEST. AN EQUIVALENT TEST TIME PARAMETER RESPONSIBLE FOR THE FUEL BURNING IS USED TO MOVE THE TIME AXIS ORIGIN TO THE LEFT FOR THE BEGINNING OF THE TEST. THE NONZERO FAILURE FRACTION AT THE REAL TEST BEGINNING IS SUPPOSED TO EXIST APPEARING AS A VERY LARGE FRACTION OF THE SPECIMENS FAILED AT THE VERY FIRST MOMENTS OF THE TEST. (FAH)
YWORDS — ISRAEL;RELIABILITY ANALYSIS;ANALYTICAL MODEL;CLADDING;STRESS CORROSION;CRACK;FAILURE

8/0/0000001-00000767/ 40

CESSION NO. — 00J0166283
TLE — TECHNICAL NOTE: LICENSEE EVENT REPORT SEQUENCE CODING AND SEARCH PROCEDURE WORKSHOP
THOR(S) — COTTRELL WB;GALLAHER RB
RPAUTH — OAK RIDGE NATIONAL LAB., TENN.
TE — 1981
PE — U
MO — 3 PPS, 2 TABS, 2 REFS, NUCLEAR SAFETY, 22(2), PP. 162-64 (MARCH-APRIL 1981)
TEGORY — 170000;230000
ITION — 0129
RP CODE — F2C
UNTRY — A
ABB — NUSA
STRACT — THE PROGRAM INCLUDED AN IN-DEPTH BRIEFING ON THE STATUS OF THE SEQUENCE CODING AND SEARCH PROCEDURE (SCSP) AND ALSO BRIEFINGS BY MEMBERS OF THE ANALYSIS AND EVALUATION OF OPERATIONAL DATA STAFF AS TO THEIR WORK AND FUTURE PLANS RELATING TO OPERATING DATA ASSESSMENT. VERY BRIEFLY, THE SCSP IS A COMPUTER-BASED RETRIEVAL SYSTEM WHICH WILL HAVE MARKEDLY IMPROVED SEARCH STRATEGY CAPABILITY FOR SUCH ITEMS AS COMMON-CAUSE FAILURES OR COMPLEX SYSTEM INTERACTION DERIVED FROM A KNOWLEDGE OF FAILURE SEQUENCES AND OTHER RELATIONSHIPS ASSOCIATED WITH AN EVENT. AN ILLUSTRATIVE EXAMPLE IS GIVEN.
YWORDS — INDUSTRY, NUCLEAR;INDUSTRY, UTILITY;AGENCY, NRC;COMPUTER PROGRAM;DATA COLLECTION;DATA PROCESSING;INFORMATION RETRIEVAL; OPERATING EXPERIENCE

3/0/0000001-00000767/ 41

CESSION NO. — 00X0166265
TLE — SPACE AND MISSILE RELIABILITY AND SAFETY PROGRAMS
RPAUTH — PICKERING RESEARCH CORP., PASADENA, CA
TE — 1981
PE — N
MO — NSAC-31 +. 295 PPS, FEB. 1981
AIL — AVAILABILITY - RESEARCH REPORTS CENTER, ELECTRIC POWER RESEARCH INST., P.O. BOX 10090, PALO ALTO, CALIF. 94303
TEGORY — 010000;230000;040000
ITION — 0129

TRY A
RACT THIS REPORT DOCUMENTS THE EVOLUTIONARY DEVELOPMENT OF RELIABILITY AND SAFETY PRACTICES IN UNMANNED AND MANNED SPACE PROJECTS, AND MILITARY SPACE AND INTERCONTINENTAL MISSILE PROJECTS, OVER THE LAST TWO DECADES. IT DISCUSSES SUCCESSES, PROBLEMS AND FAILURES, AND IDENTIFIES THE CORRECTIVE ACTIONS WHICH RESULTED IN A DRAMATIC INCREASE IN RELIABILITY, EVEN THOUGH THE COMPLEXITY AND MISSION REQUIREMENTS OF THESE PROJECTS GREATLY INCREASED WITH TIME. THE REPORT DISTILLS THE "LESSONS LEARNED" AND PRESENTS THEM IN A FASHION WHICH CAN BE USEFUL TO OTHER ORGANIZATIONS RESPONSIBLE FOR THE DEVELOPMENT, IMPLEMENTATION AND OPERATION OF EXTREMELY COMPLEX AND TECHNICALLY ADVANCED SYSTEMS WHICH MUST HAVE HIGH RELIABILITY.
YWORDS SPACECRAFT;RELIABILITY, COMPONENT;RELIABILITY, SYSTEM;FAULT TREE ANALYSIS;SAFETY PROGRAM;TESTING;TRAINING;REDUNDANCE;HUMAN FACTORS;RELIABILITY ANALYSIS;EPRI

E/0/0000001-0000076//   42
CESSION NO. 00J0166284
TLE WASH-1400: A COMPARISON OF EXPERIENCE AND PREDICTION
THOR(S) LELLOUCHE GS
RPAUTH ELECTRIC POWER RESEARCH INST., PALO ALTO, CA
TE 1981
PE J
MO 4 PPS, 4 FIGS, 5 REFS, NUCLEAR TECHNOLOGY, 53(2), PP. 231-4 (MAY 1981)
TEGORY 230000
ITION 0129
RP CODE EPR
UNTRY A
ABS NUAT
STRACT IN AN ELECTRIC POWER RESEARCH INSTITUTE REPORT, IT WAS CONCLUDED THAT THE ERROR BOUNDS IN WASH-1400, THE REACTOR SAFETY STUDY, ALTHOUGH PERHAPS UNDERSTATED, ARE NOT NECESSARILY "GREATLY UNDERSTATED," AS CLAIMED BY THE LEWIS COMMITTEE. THE RELATIONSHIP BETWEEN AN EXPERIMENTAL DATA BASE FOR OPERATING LIGHTWATER REACTORS (LWRS) AND THE PREDICTIONS OF WASH-1400 HAVE BEEN EXAMINED. THE DATA BASE CONSISTS OF: 1) U.S. COMMERCIAL LWR REACTORS, 2) WORLD LWR REACTORS, AND 3) WORLD LWR PLUS U.S. NAVY REACTORS. ESTIMATIONS OF AN EXPECTED INCREASE IN EXPERIENCE INDICATE THAT THE WASH-1400 CORE-MELT PROBABILITY CANNOT BE AN UNDERESTIMATION. THE MAXIMUM DEGREE OF UNQUANTIFIED PROBABILITY IN THE WASH-1400 CALCULATION COMPARED TO EXPERIENCE IS THAT OF THE MARGIN. THE INCREASE IN WASH-1400 UNCERTAINTY CANNOT BE GREATER THAN A FACTOR OF 4, AND THE WASH-1400 MEDIAN CANNOT BE LOW BY MORE THAN A FACTOR OF 4.
YWORDS PREDICTION;SAFETY MARGIN;REACTOR, LWR;DATA COLLECTION; PROBABILITY;CORE MELTDOWN

8/0/0000001-0000078//   43
CESSION NO. 00E0165711
TLE RELIABILITY ANALYSIS OF THE SHUTDOWN HEAT REMOVAL SYSTEM FOR THE CONCEPTUAL DESIGN STUDY
THOR(S) BRYAN TL;SIMONELLI RB
RPAUTH WESTINGHOUSE ELECTRIC CORP., MADISON, PA.
TE 1980
PE H
MO WARD-SR-94000-17 +. 113 PPS, 7 TABS, REFS, SEPT. 1980
AIL AVAILABILITY - LIMITATIONS ON DISTRIBUTION; SEND REQUESTS TO DOE TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830
TEGORY 230000;120000;050000
ITION 0128
RP CODE WES
UNTRY A
STRACT SHUTDOWN HEAT REMOVAL SYSTEM (SHRS) RELIABILITY ASSESSMENT WAS PRIMARILY PERFORMED IN SUPPORT OF THE CONCEPTUAL DESIGN STUDY (CDS) PHASE II EFFORTS. IT IS A DETAILED EXAMINATION OF THE DRACS-IRACS DEDICATED HEAT REMOVAL COMBINATION. THE COMBINATION SELECTED FROM THE PHASE I TRADE-OFF STUDIES OF VIABLE HEAT REMOVAL SYSTEMS. USING METHODOLOGY CONSISTENT WITH THAT USED DURING CDS PHASE I, THE ANNUAL PROBABILITY OF SHUTDOWN HEAT REMOVAL SYSTEM FAILURE WAS CALCULATED AND COMPARED WITH ESTABLISHED GOALS. THIS STUDY CONCLUDES THAT THE DRACS-IRACS

SHRS COMBINATION EXCEEDS, WITH CONSIDERABLE MARGIN, THE PRESENT SHRS RELIABILITY GOALS.

ORDS SAFETY ANALYSIS;REACTOR SHUTDOWN;HEAT TRANSFER;CONDENSER COOLING SYSTEM;CIRCULATION, NATURAL;REACTOR, POWER;TEST, SYSTEM OPERABILITY;RELIABILITY ANALYSIS;SHUTDOWN COOLING SYSTEM; FAILURE MODE ANALYSIS;ACCIDENT, LOSS OF POWER

8/07/0000001-00000767/ 44

CESSION NO. 00E0165697

TLE INTEGRATION OF NDE RELIABILITY AND FRACTURE MECHANICS PHASE I REPORT

THOR(S) BECKER FL;DOCTOR SR;SELBY GP

RPAUTH PACIFIC NORTHWEST LAB., RICHLAND, WA

TE 1980

PE H

MO NUREG/CR 1696 (VOL. 1) + PNL-3469 +. 220 PPS. FIGS. REFS. OCT. 1980

AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

TEGORY 110000;230000

ITION 0128

RP CODE BNR

UNTRY A

STRACT A SURVEY WAS MADE OF THE STATE OF PRACTICE FOR ULTRASONIC ISI OF LWR PRIMARY SYSTEM PIPING WELDS. FRACTURE MECHANICS CALCULATIONS WERE MADE TO ESTABLISH REQUIRED NONDESTRUCTIVE TESTING SENSITIVITIES. IT WAS FOUND THAT FATIGUE FLAWS LESS THAN 25% OF WALL THICKNESS WOULD NOT GROW TO FAILURE WITHIN AN INSPECTION INTERVAL OF 10 YEARS. HOWEVER, IN SOME CASES FAILURE COULD OCCUR CONSIDERABLY FASTER. STATISTICAL METHODS FOR PREDICTING AND MEASURING THE EFFECTIVENESS AND RELIABILITY OF ISI WERE DEVELOPED AND WILL BE APPLIED IN THE "ROUND ROBIN INSPECTIONS" OF PHASE II. MEASUREMENTS WERE MADE OF THE INFLUENCE OF FLAW CHARACTERISTICS (I.E., ROUGHNESS, TIGHTNESS, AND ORIENTATION) ON INSPECTION RELIABILITY. THESE MEASUREMENTS, AS WELL AS THE PREDICTIONS OF A STATISTICAL MODEL FOR INSPECTION RELIABILITY, INDICATE THAT CURRENT REPORTING AND RECORDING SENSITIVITIES ARE ADEQUATE. (FAH)

YWORDS R AND D PROGRAM;ULTRASONICS;TEST, NONDESTRUCTIVE;RELIABILITY ANALYSIS;FRACTURE TOUGHNESS;STATISTICAL ANALYSIS;PIPES AND PIPE FITTINGS;WELDS;HJCK;NRC-5

8/07/0000001-00000767/ 45

CESSION NO. 00E0165636

TLE FRANTIC II - A COMPUTER CODE FOR TIME DEPENDENT UNAVAILABILITY ANALYSIS

THOR(S) VESELY WE;DICKEY JM;HALL RE

RPAUTH BROOKHAVEN NATIONAL LAB., UPTON, N.Y.

TE 1981

PE H

MO NUREG/CR-1924 + BNL-NUREG-51355 +. 90 PPS. TABS. FIGS. APRIL 1981

AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

TEGORY 170000;230000

ITION 0127

RP CODE BZA

UNTRY A

STRACT THE FRANTIC COMPUTER CODE EVALUATES THE TIME DEPENDENT AND AVERAGE UNAVAILABILITY FOR ANY GENERAL SYSTEM MODEL. THE CODE IS WRITTEN IN FORTRAN IV FOR THE CDC 7600 COMPUTER. NON-REPAIRABLE COMPONENTS, MONITORED COMPONENTS, AND PERIODICALLY TESTED COMPONENTS ARE HANDLED. ONE UNIQUE FEATURE OF FRANTIC IS THE DETAILED, TIME DEPENDENT MODELING OF PERIODIC TESTING WHICH INCLUDES THE EFFECTS OF TEST DOWNTIMES, TEST OVERRIDES, DETECTION INEFFICIENCIES, AND TEST-CAUSED FAILURES. HUMAN ERRORS AND COMMON MODE FAILURES CAN BE INCLUDED BY ASSIGNING AN APPROPRIATE CONSTANT PROBABILITY FOR THE CONTRIBUTORS.

YWORDS COMPUTER PROGRAM;AVAILABILITY;COMPONENTS;FAILURE, COMPONENT; SENSITIVITY ANALYSIS;RELIABILITY ANALYSIS;TESTING;HUMAN FACTORS; FAILURE, COMMON MODE;HJCK;NRC-RG;FAILURE, OPERATOR ERROR

SSION NO. 00X0165569
LE ANALYSIS OF FUEL ROD BEHAVIOR WITHIN A ROD BUNDLE OF A PRESSURIZED WATER REACTOR UNDER THE CONDITIONS OF A LOSS OF COOLANT ACCIDENT (LOCA) USING PROBABILISTIC METHODOLOGY (IN GERMAN)
THOR(S) SENGPIEL W
RPAUTH KERNFORSCHUNGSZENTRUM KARLSRUHE, F.R. GERMANY
TE 1980
PE N
MO KFK-2965 + GERRSR-701 +. 111 PPS, FIGS, REFS, DEC. 1980
NGUAGE GERMAN
AIL AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
TEGORY 120000;050000;230000
ITION 0127
RP CODE KSR
UNTRY G
STRACT AN ANALYTICAL MODEL IS APPLIED TO STUDY THE BEHAVIOR OF FUEL RODS UNDER ACCIDENT CONDITIONS FOLLOWING THE DOUBLE-ENDED PIPE RUPTURE BETWEEN COOLANT PUMP AND PRESSURE VESSEL IN THE PRIMARY SYSTEM OF A 1300 MW(EL)-PWR. SPECIFICALLY A ROD BUNDLE IS CONSIDERED CONSISTING OF 236 FUEL RODS, THAT IS SUBJECTED TO SEVERE THERMAL AND MECHANICAL LOADING. THE RESULTS OBTAINED INDICATE THAT PLASTIC CLAD DEFORMATIONS WITH CIRCUMFERENTIAL CLAD STRAINS OF MORE THAN 30% CANNOT BE EXCLUDED FOR HOT RODS OF THE REFERENCE BUNDLE. HOWEVER COPLANAR COOLANT CHANNEL BLOCKAGES OF SIGNIFICANT EXTENT SEEM TO BE PROBABLE WITHIN THAT BUNDLE ONLY UNDER CERTAIN BOUNDARY CONDITIONS WHICH ARE ASSUMED TO BE PESSIMISTIC. (FAH)
YWORDS GERMANY;REACTOR, PWR;FUEL ROD;BEHAVIOR;ACCIDENT, LOSS OF COOLANT;PROBABILITY;ANALYTICAL MODEL;FOREIGN EXCHANGE

8707/0000001-00000767/ 47
CESSION NO. 00X0165338
TLE INTEGRATING RELIABILITY ANALYSIS AND DESIGN
THOR(S) RASMUSON DM
RPAUTH EG&G IDAHO INC., IDAHO FALLS
TE 1980
PE N
MO ALO-131 + EGG-IS-5187 +. 65 PPS, FIGS, OCT. 1980
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 010000;180000;230000
ITION 0127
RP CODE EGG
UNTRY A
STRACT THIS REPORT DESCRIBES THE INTERACTIVE RELIABILITY ANALYSIS PROJECT AND DEMONSTRATES THE ADVANTAGES OF USING COMPUTER-AIDED DESIGN SYSTEMS (CADS) IN RELIABILITY ANALYSIS. COMMON CAUSE FAILURE PROBLEMS REQUIRE PRESENTATIONS OF SYSTEMS, ANALYSIS OF FAULT TREES, AND EVALUATION OF SOLUTIONS TO THESE. RESULTS HAVE TO BE COMMUNICATED BETWEEN THE RELIABILITY ANALYST AND THE SYSTEM DESIGNER. USING A COMPUTER-AIDED DESIGN SYSTEM SAVES TIME AND MONEY IN THE ANALYSIS OF DESIGN. COMPUTER-AIDED DESIGN SYSTEMS LEND THEMSELVES TO CABLE ROUTING, VALVE AND SWITCH LISTS, PIPE ROUTING, AND OTHER COMPONENT STUDIES. AT EG&G IDAHO, INC., THE APPLICON CADS IS BEING APPLIED TO THE STUDY OF WATER REACTOR SAFETY SYSTEMS.
YWORDS RELIABILITY ANALYSIS;DESIGN;FAULT TREE ANALYSIS;FAILURE, COMMON MODE;LPCI;REACTOR, LWR

8707/0000001-00000767/ 48
CESSION NO. 00X0165234
TLE THE EXPERIMENTAL FACILITY FOR CONTAINMENT SUMP RELIABILITY STUDIES (GENERIC TASK A-43)
THOR(S) DURGIN WW;JANIK CR;PADMANABHAN M
RPAUTH WORCESTER POLYTECHNIC INST., HOLDEN, MASS.
TE 1980
PE N
MO ALO-132 + ARL-120-80/M398 +. 64 PPS, 39 FIGS, 6 REFS, DEC. 1980
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161

GORY 110000;230000
TION 0127
ITRY A
TRACT THIS REPORT DESCRIBES THE TEST FACILITY CONSTRUCTED TO CONDUCT TESTS ON UNRESOLVED SAFETY ISSUES ASSOCIATED WITH CONTAINMENT SUMP PERFORMANCE DURING THE RECIRCULATION MODE. THE TEST FACILITY CONSISTS OF A MAIN TANK WITH SUMP, SUCTION PIPES WITH VARIABLE DIAMETERS AND POSITIONS, A PUMP PIT TANK, AND ASSOCIATED PIPING FOR THE SIMULATION OF BREAK AND DRAIN FLOWS. SUMP PERFORMANCE IS DETERMINED THROUGH THE OBSERVATION OF VORTEX FORMATION IN THE MAIN TANK AND THE MEASUREMENT OF SWIRL, PRESSURE GRADIENT, AND ENTRAINED AIR IN THE SUCTION PIPES. A SOPHISTICATED DATA ACQUISITION SYSTEM, WITH COMPUTER INTERFACE, ALLOWS THE TEST FLOW PARAMETERS TO BE SET AND TEST DATA TO BE TAKEN (WITH THE EXCEPTION OF VORTEX OBSERVATIONS) FROM A SINGLE CENTRAL OFFICE. (FAH)
EYWORDS CONTAINMENT SUMP;FLOW;FLOW, VORTEX;FLOW THEORY AND EXPERIMENTS; PIPES AND PIPE FITTINGS;PUMPS;TESTING

87070000001-00000767/ 49
CCESSION NO. 0GE0165118
ITLE PALO VERDE NUCLEAR GENERATING STATION AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS
ORPAUTH ARIZONA PUBLIC SERVICE CO., PHOENIX, AZ
ATE 1981
YPE H
EMO APPROX. 160 PGS. LTR W/ATTACH. TO NRC DIRECTOR OF NUCLEAR REACTOR REGULATION, FEB 10, 1981. DOCKETS 50-528/529/530. TYPE--PWR, MFG--CE, AE--BECH. DCS NO. 8102250251
VAIL AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (05 CENTS/PAGE -- MINIMUM CHARGE $2.00)
ATEGORY 050000;170000;230000
DITION 0126
ORP CODE ARS
OUNTRY A
BSTRACT THE PALO VERDE AFS WAS ANALYZED CONSIDERING VARIOUS DESIGN ALTERNATIVES AS A RESULT OF THE CONCERNS EXPRESSED BY THE NRC. THIS STUDY CONSIDERED FOUR DIFFERENT DESIGN ALTERNATIVES. THE FIRST ONE IS THE CURRENT DESIGN WHICH CONSISTS OF ONE TURBINE-DRIVEN EMERGENCY FEEDWATER TRAIN, ONE MOTOR-DRIVEN EMERGENCY FEEDWATER TRAIN, AND A MANUAL START NON IE AC POWER MOTOR-DRIVEN AUXILIARY FEEDWATER TRAIN. IN CASE 2 THE STARTUP AUXILIARY FEEDWATER PUMP WAS GIVEN THE CAPABILITY OF BEING POWERED FROM THE TRAIN A DIESEL GENERATOR BY MANUAL START. CASE 2A IS THE SAME AS CASE 2 EXCEPT AUTOMATIC START IS PROVIDED. IN CASE 3, A FOURTH FEEDWATER PUMP TRAIN WAS ADDED.
EYWORDS PALO VERDE 1 (PWR);PALO VERDE 2 (PWR);PALO VERDE 3 (PWR); REACTOR, PWR,RELIABILITY ANALYSIS;AUXILIARY;FEEDWATER;DESIGN STUDY;COOLING SYSTEM, SECONDARY

87070000001-00000767/ 50
CCESSION NO. 0000165107
ITLE A METHODOLOGY FOR EVALUATING THE PROBABILITY FOR FIRE LOSS OF NUCLEAR POWER PLANT SAFETY FUNCTIONS
UTHOR(S) GALLUCCI RHV
ORPAUTH RENSSELAER POLYTECHNIC INST., TROY, N.Y.
ATE 1980
YPE U
EMO 407 PPS, 1980 (THESIS)
VAIL AVAILABILITY - DISSERTATION COPIES, UNIVERSITY MICROFILMS INTERNATIONAL, P.O. BOX 1764, ANNA ARBOR, MICHIGAN 48106 (ORDER NO. 8020404)
ATEGORY 010000;120000;230000
DITION 0126
ORP CODE MCZ
OUNTRY A
BSTRACT METHODOLOGY HAS BEEN DEVELOPED FOR EVALUATION OF PROBABILITY FOR LOSS OF NUCLEAR POWER PLANT SAFETY FUNCTIONS DUE TO FIRES. FRAMEWORK FOR INVESTIGATION OF FIRE SCENARIOS INVOLVING SAFETY-RELATED EQUIPMENT HAS BEEN ESTABLISHED WHICH MODELS FIRE DEVELOPMENT AS A SERIES OF IGNITION, DETECTION, SUPPRESSION, AND PROPAGATION STEPS. IN ADDITION TO PRESENTING VARIOUS MODELS FOR QUANTITATIVE EVALUATION OF PROBABILITIES FOR THESE

STEPS. GENERIC VALUES HAVE BEEN CALCULATED TO ILLUSTRATE APPLICATION OF THIS METHODOLOGY TO AN EXISTING BWR. TO PLACE THE NUMERICAL RESULTS IN CONTEXT OF REACTOR ACCIDENT CONSEQUENCES. PROBABILITY FOR CORE DAMAGE DUE TO A NUCLEAR PLANT FIRE AT THIS REPRESENTATIVE BWR HAS BEEN ESTIMATED. VALUE OF 2.1 X 10(-4)/PLANT-YEAR WAS CALCULATED, WITH AN UPPER BOUND OF .0013/PLANT-YEAR. FIRE IGNITION, AUTOMATIC AND HUMAN DETECTION, EXTINGUISHING AGENT EFFECTIVENESS, AND FIRE PROPAGATION HAVE BEEN DETERMINED TO BE THE DOMINANT FACTORS FOR CONSIDERATION IN THE ANALYSIS OF A FIRE SCENARIO.

YWORDS   POWER PLANT, NUCLEAR;PROBABILITY;FIRE;IGNITION;REACTOR, BWR; ACCIDENT MODEL;FIRE PROTECTION

870/0000001-00000767/   51

CCESSION NO.   00X0164992

TLE   NUMERICAL INVESTIGATION OF THE LONG-TIME TEMPERATURE BEHAVIOR OF A PARTIALLY UNCOVERED REACTOR CORE

THOR(S)   BRACHT K

RPAUTH   GESELLSCHAFT FUR REAKTORSICHERHEIT (GRS) MBH, F.R. GERMANY

TE   1980

PE   N

MO   GRS-A-460 + GRRSR-598 +. 127 PPS. FIGS. JUNE 1980

NGUAGE   GERMAN

AIL   AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.

TEGORY   050000;230000

ITION   0126

RP CODE   GRS

UNTRY   G

STRACT   IN THE CASE OF DEFICIENT LWR SAFETY SYSTEMS THE RISK OF A CORE MELT DOWN ACCIDENT IS GIVEN. FOR A FURTHER QUANTIFICATION OF THIS RISK IT IS NECESSARY TO KNOW THE LIMITING CONDITIONS UNDER WHICH A CORE REMAINS COOLABLE AND A MELT DOWN ACCIDENT CAN BE PREVENTED. THE AIM OF THE PRESENT REPORT IS TO CONTRIBUTE TO THE DETERMINATION OF THESE CONDITIONS. IN THIS REPORT THE TIME DEPENDENT DEVELOPMENT OF THE LOCAL TEMPERATURES AND CLADDING OXIDATION IN A CORE IS CALCULATED FOR THE CASE OF A POST LOCA SYSTEM FAILURE LEADING TO A SITUATION WHERE A LWR-CORE IS PARTIALLY UNCOVERED FOR A LONGER TIME PERIOD.

YWORDS   REACTOR, LWR;ACCIDENT, LOSS OF COOLANT;CHEMICAL REACTION;CORE MELTDOWN;COMPUTER PROGRAM;NUMERICAL METHOD;EMERGENCY COOLING; TEMPERATURE;FLOW, TWO PHASE;FOREIGN EXCHANGE

870/0000001-00000767/   52

CESSION NO.   00C0164817

TLE   SAFETY AND RELIABILITY ASSESSMENT. PROCEEDINGS OF THE COURSE HELD AT THE COUNCIL FOR SCIENTIFIC AND INDUSTRIAL RESEARCH CONFERENCE CENTRE, SCIENTIA, PRETORIA, 9-20 JULY 1979. (CONF., N.DATA) (C52 C53 E22)

RPAUTH   ATOMIC ENERGY BOARD, PRETORIA, SOUTH AFRICA

TE   1979

PE   L

MO   567 PPS. 1979

AIL   AVAILABILITY - ADMINISTRATIVE OFFICER, LICENSING BRANCH, ATOMIC ENERGY BOARD, PRIVATE BAG X256, PRETORIA, 0001 AT SA R40

TEGORY   010000;090000;210000;150000;230000;180000

ITION   0126

UNTRY   X

STRACT   THIS REPORT CONTAINS THE PAPERS FROM THIS CONFERENCE. THE FOLLOWING TOPICS WERE DISCUSSED: SAFETY STANDARDS; LICENSING; BIOLOGICAL EFFECTS OF RADIATION; WHAT IS A PWR; SAFETY PRINCIPLES IN THE DESIGN OF A NUCLEAR REACTOR; RADIO-RELEASE ANALYSIS; QUALITY ASSURANCE; THE STAFFING, ORGANIZATION AND TRAINING FOR A NUCLEAR POWER PLANT PROJECT; EVENT TREES, FAULT TREES AND PROBABILITY; AUTOMATIC PROTECTIVE SYSTEMS; SOURCES OF FAILURE-RATE DATA; INTERPRETATION OF FAILURE DATA; SYNTHESIS AND RELIABILITY; QUANTIFICATION OF HUMAN ERROR IN MAN-MACHINE SYSTEMS; DISPERSION OF NOXIOUS SUBSTANCES THROUGH THE ATMOSPHERE; CRITICALITY ASPECTS OF ENRICHMENT AND RECOVERY PLANTS; AND RISK AND HAZARD ANALYSIS. EXTENSIVE EXAMPLES ARE GIVEN AS WELL AS CASE STUDIES. (EWH)

YWORDS   RELIABILITY ANALYSIS;SAFETY EVALUATION;CODES AND STANDARDS; RADIATION EFFECT, ECOSYSTEM;HUMAN FACTORS;RADIATION SAFETY AND

CONTROL;FAULT TREE ANALYSIS;RADIOACTIVITY RELEASE;LICENSING PROCESS

/07/0000001-0000076// 53

CESSION NO.	00X0163953
TLE	PROBCALC USER'S MANUAL: A COMPUTER PROGRAM FOR PROBABILITY CALCULATIONS
THOR(S)	INGRAM GE;ELLERATH JG
RPAUTH	GENERAL ELECTRIC CO., SUNNYVALE, CALIF.
ATE	1980
YPE	N
EMO	GEFR-00408 +. 65 PPS. 14 TABS. 18 FIGS. AUG. 1980
VAIL	AVAILABILITY - LIMITATIONS ON DISTRIBUTION; SEND REQUESTS TO DOE TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830
ATEGORY	230000
DITION	0124
ORP CODE	GEC
OUNTRY	A
BSTRACT	THIS USER'S MANUAL DOCUMENTS THE MATHEMATIC BASIS FOR, AND EXPLAINS THE USE OF THE COMPUTER CODE PROBCALC. PROBCALC IS A GE-TIMESHARE COMPUTER PROGRAM, WRITTEN IN FORTRAN, WHICH CALCULATES PROBABILITIES OF DISCRETE STATE LOGIC NETWORKS SUCH AS FAULT-TREES AND PROBABILITY BLOCK DIAGRAMS (PBDS). THE MAJOR FEATURE OF THE PROGRAM IS ITS ABILITY TO CALCULATE PROBABILITIES OF MODELS HAVING DEPENDENT EVENTS. THESE DEPENDENCIES RESULT FROM THE REPRESENTATION OF A SINGLE PHYSICAL EVENT IN MORE THAN ONE PLACE IN THE MODEL LOGIC. THIS MANUAL IS ORIENTED TOWARDS PBDS BUT FAULT-TREES CAN BE EVALUATED DIRECTLY BY KNOWING THAT AN "AND" GATE IS SYNONYMOUS WITH "SERIAL" BLOCKS AND AN "OR" GATE IS SYNONYMOUS WITH "PARALLEL" BLOCKS.
EYWORDS	PROBABILITY;FAULT TREE ANALYSIS;SENSITIVITY ANALYSIS;COMPUTER PROGRAM;ACCIDENT, PROBABILITY OF

1 8/07/0000001-0000076// 54

CCESSION NO.	00C0163552
ITLE	AN ACCEPTABLE FUTURE NUCLEAR ENERGY SYSTEM
UTHOR(S)	OHANIAN MJ;FIREBAUGH MW
ORPAUTH	INST. FOR ENERGY ANALYSIS, OAK RIDGE, TENN.
ATE	1979
YPE	L
EMO	ORAU/IEA-80-3(P) +. 250 PPS. FROM GATLINBURG II. CONDENSED WORKSHOP PROCEEDINGS; GATLINBURG, TENN., DEC. 10-12, 1979
VAIL	AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
ATEGORY	010000;170000;180000;230000
DITION	0123
ORP CODE	IEA
OUNTRY	A
BSTRACT	THIS VOLUME SUMMARIZES THE PROCEEDINGS OF THE SECOND GATLINBURG WORKSHOP ON AN ACCEPTABLE NUCLEAR ENERGY SYSTEM. THIS WORKSHOP WAS A SEQUEL TO A SIMILAR ONE HELD IN GATLINBURG, TENNESSEE, IN DECEMBER 1976. BOTH WERE SPONSORED BY THE DEPARTMENT OF ENERGY AND WERE CONVENED BY THE INSTITUTE FOR ENERGY ANALYSIS (IEA) OF OAK RIDGE ASSOCIATED UNIVERSITIES.
EYWORDS	N-POWER FORECAST;N-POWER, SAFETY OF;ENERGY SOURCE;ENERGY POLICY;INCIDENT;REACTOR, PWR;SOCIO/PHILOSOPHICAL CONSIDERATION;SAFETY EVALUATION;SITING;THREE MILE ISLAND 2 (PWR)

8/07/0000001-0000076// 55

CESSION NO.	00X0162922
TLE	COMPUTER PROGRAM ZUSTA FOR RELIABILITY AND AVAILABILITY OF NUCLEAR POWER PLANTS USING THE METHOD OF STATE ANALYSIS (IN GERMAN)
THOR(S)	CYR W;WEHLING J;KRETZEN H-H
RPAUTH	INTERATOM GMBH, F.R. GERMANY
TE	1980
PE	N
MO	INTAT 3244445 + GERRSR-560 +. 225 PPS. FIGS. JAN. 1980
NGUAGE	GERMAN
AIL	AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.

GORY   010000;090000;230000
TION   0122
NTRY   G
TRACT   THE MONTE-CARLO CODE ZUSTA SIMULATES THE BEHAVIOR OF A NUCLEAR POWER PLANT DURING NORMAL OPERATION AS WELL AS ACCIDENTS FOR A FIXED PERIOD. THE CODE IS WRITTEN IN FORTRAN EXTENDED VERSION 4 FOR THE CD-CYBER 172 COMPUTER. THE REALIZATIONS OF PLANT OPERATION SIMULATED BY ZUSTA ARE SUBJECT TO A STATISTICAL EVALUATION, YIELDING STATE-SPECIFIC QUANTITIES LIKE FREQUENCY OF OCCURRENCE, MEAN WAIT AND STATE-PROBABILITY AS WELL AS THE DISTRIBUTION OF THE ANNUAL (RADIOACTIVE) RELEASE AS AN OVERALL RISK-RELATED RESULT.
EYWORDS   GERMANY;MONTE CARLO;COMPUTER PROGRAM;FAULT TREE ANALYSIS; AVAILABILITY;POWER PLANT, NUCLEAR;REACTOR, FAST;FOREIGN EXCHANGE;RELIABILITY ANALYSIS

8/07/0000001-0000076//   56
CCESSION NO.   00C0162896
ITLE   SOLVING RELIABILITY MODELS OF NUCLEAR SYSTEMS
UTHOR(S)   DOYON LR
ATE   1977
YPE   L
EMO   10 PPS. PROCEEDINGS 1977 ANNUAL RELIABILITY & MAINTAINABILITY SYMPOSIUM, PP. 322-31 (JAN. 18, 1977) (PAPER 1460, 77RM056)
ATEGORY   090000;230000
DITION   0122
OUNTRY   A
BSTRACT   IN THIS INVESTIGATIVE STUDY THE AUTHOR DERIVES A MARKOVIAN STATE-TRANSITION DIAGRAM FOR THE RELIABILITY/AVAILABILITY/MAINTAINABILITY MODEL OF A TYPICAL COOLANT-WATER SYSTEM FOR A NUCLEAR-REACTOR POWER-GENERATION PLANT. THIS SYSTEM HAS A CONSIDERABLE AMOUNT OF REDUNDANCY IN EQUIPMENT. THE TIMES-BETWEEN-FAILURE OBEY THE EXPONENTIAL PROBABILITY LAW FOR ELECTRONIC EQUIPMENTS AND THE NORMAL PROBABILITY LAW FOR MECHANICAL APPARATUS. THE TIMES-TO-REPAIR OBEY THE LOGNORMAL PROBABILITY LAW FOR ELECTRONIC EQUIPMENTS AND THE NORMAL PROBABILITY LAW FOR MECHANICAL EQUIPMENTS. FOR SOLVING THE MODEL ANALYTICALLY, THE AUTHOR OFFERS THE USE OF THE COMPUTER ALGORITHM AFARS (ALGOTITHM FOR ANALYZING THE RELIABILITY OF SYSTEMS) HE DEVISED. FOR SOLVING THE MODEL BY COMPUTER SIMULATION, HE RECOMMENDS THE USE OF A SIMULATION PROGRAM SUCH AS GERTS, GASP, OR GPSS. (EWH)
EYWORDS   RELIABILITY ANALYSIS;RELIABILITY, SYSTEM;THEORETICAL INVESTIGATION;ANALYTICAL MODEL;COMPUTER PROGRAM

8/07/0000001-0000076//   57
CCESSION NO.   00C0162895
ITLE   A PROBABILISTIC APPROACH TO DESIGN FOR THE ECCS OF A PWR
UTHOR(S)   GACHOT B
ATE   1977
YPE   L
EMO   11 PPS. PROCEEDINGS 1977 ANNUAL RELIABILITY & MAINTAINABILITY SYMPOSIUM, PP. 332-42 (JAN. 18, 1977)(PAPER 1461, 77RM057)
ATEGORY   090000;230000
DITION   0122
OUNTRY   A
BSTRACT   IN THIS STUDY, A FAULT-TREE ANALYSIS AND A PROBABILISTIC EVALUATION ARE USED TO COMPARE THE DIFFERENT TYPES OF DESIGNS, BASED ON A DIFFERENT ORDER OF REDUNDANCY FOR THE EMERGENCY CORE COOLING SYSTEM OF A 1300 MW PRESSURIZED WATER REACTOR. THE EFFECTS OF MAINTENANCE AND COMMON-CAUSE FAILURES ARE TAKEN INTO ACCOUNT AND APPEAR TO BE MORE AND MORE IMPORTANT WHEN THE SYSTEM REDUNDANCY INCREASES. (EWH)
YWORDS   RELIABILITY ANALYSIS;RELIABILITY, SYSTEM;PROBABILITY;REACTOR, PWR;EMERGENCY COOLING SYSTEM,FAULT TREE ANALYSIS;COMPARISON; DESIGN STUDY;FAILURE, COMMON MODE

8/07/0000001-0000076//   58
CESSION NO.   00C0162894
TLE   NUCLEAR POWER PLANT RELIABILITY AUDITS
THOR(S)   ESSINGER T
TE   1977
PE   L
MO   3 PPS. PROCEEDINGS 1977 ANNUAL RELIABILITY & MAINTAINABILITY

SYMPOSIUM, PP. 343-345 (JAN. 18, 1977)(PAPER 1462, 77RM058)
GORY 090000;230000
TION 0122
NTRY A
TRACT IMPROVING NUCLEAR POWER PLANT RELIABILITY IS ONE OF THE MOST FRUITFUL WAYS OF OPTIMIZING A UTILITY'S RETURN ON INVESTMENT. CURRENTLY, FLORIDA POWER & LIGHT COMPANY IS EXTRACTING USEFUL INFORMATION FROM NUMEROUS SOURCES THROUGH RELIABILITY AUDITS TO INCREASE PLANT RELIABILITY. THROUGH AN ORGANIZED MANAGEMENT APPROACH, VARIOUS TECHNICAL DISCIPLINES JOIN FORCES TO GATHER PERTINENT INFORMATION, BOTH SPECIFIC AND GENERIC, CONCERNING THE FAILED COMPONENT, ITS MAINTENANCE HISTORY, AND ITS SUPPLIER. AFTER THE TEAM ANALYZES THE DATA, IT MAKES RECOMMENDATIONS FOR CORRECTIVE ACTION TO THOSE CAPABLE OF ACHIEVING IT. (EWH)
YWORDS RELIABILITY ANALYSIS;RELIABILITY, COMPONENT;AVAILABILITY;POWER PLANT, NUCLEAR;QUALITY ASSURANCE

870/0000001-00000767/ 59
CESSION NO. 00C0162893
TLE RELIABILITY ASSESSMENT FOR HEAVY MACHINERY BY "HI-FMECA" METHOD
THOR(S) NUKADA K;MIKI M;ONODERA
ATE 1977
YPE L
EMO 7 PPS, PROCEEDINGS 1977 ANNUAL RELIABILITY & MAINTAINABILITY SYMPOSIUM, PP. 345-52 (JAN. 18, 1977)(PAPER 1463, 77RM059)
ATEGORY 390000;100000;230000
DITION 0122
OUNTRY A
BSTRACT THE "HI-FMECA" STANDS FOR HITACHI-FAILURE MODE EFFECT AND CRITICALITY ANALYSIS. AT HITACHI WORKS OF HITACHI, LTD. IN JAPAN, THIS RELIABILITY ASSESSMENT METHOD ("HI-FMECA") HAS BEEN USED AND PROVEN TO BE HELPFUL IN DESIGNING AND ANALYZING THE DESIGN OF HEAVY EQUIPMENT FOR POWER GENERATING STATIONS, THE SAME TYPE OR MODEL OF WHICH IS SELDOM REPRODUCED. THIS PAPER WILL PRESENT THE "HI-FMECA" METHOD BY MEANS OF THE EXAMPLES FOR AN AUXILIARY SYSTEM OF A NUCLEAR POWER GENERATING PLANT. (EWH)
YWORDS RELIABILITY ANALYSIS;RELIABILITY, COMPONENT;FAILURE, EQUIPMENT; FAILURE MODE ANALYSIS;JAPAN;ANALYTICAL TECHNIQUE;ELECTRIC POWER, AUXILIARY;GENERATOR, DIESEL

870/0000001-00000767/ 60
CESSION NO. 00C0162892
TLE SYSTEM PROBABILISTIC STUDIES AT THE NUCLEAR REGULATORY COMMISSION
THOR(S) VESELY WE;PITTMAN JW
TE 1977
PE L
MO 2 PPS, PROCEEDINGS 1977 ANNUAL RELIABILITY & MAINTAINABILITY SYMPOSIUM, PP. 320-21 (JAN. 18, 1977)(PAPER 1459, 77RM055)
TEGORY 090000;230000
ITION 0122
UNTRY A
STRACT THIS PAPER BRIEFLY HIGHLIGHTS SOME OF THE WORK IN PROBABILISTIC ANALYSES BEING CARRIED OUT AT THE NUCLEAR REGULATORY COMMISSION. THE PARTICULAR WORK DESCRIBED, COVERING SOME OF THOSE AREAS IN WHICH THE AUTHORS ARE DIRECTLY INVOLVED, IS BROKEN INTO SYSTEMS EVALUATION STUDIES, HUMAN ERROR AND SYSTEM QUANTIFICATION TECHNIQUES, AND COMPONENT DATA ANALYSIS STUDIES. (EWH)
YWORDS RELIABILITY ANALYSIS;PROBABILITY;AGENCY, NRC;RELIABILITY, SYSTEM;DESIGN STUDY;ANALYTICAL TECHNIQUE;FAULT TREE ANALYSIS

870/0000001-00000767/ 61
CESSION NO. 00J0162490
TLE REPORT TO THE PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND: TECHNICAL ASSESSMENT TASK FORCE REPORTS VOL. I: SUMMARY
THOR(S) JAFFE L
TE 1980
PE D
MO 90 PPS, FIGS, REFS, PROGRESS IN NUCLEAR ENERGY, 6(1-3), PP. 1-90 (1980)
TEGORY 170000;120000;010000;050000;230000

ION 0122
TRY U
RACT CONTAINS: SUMMARY SEQUENCE OF EVENTS AND ANALYSES OF CORE DAMAGE, THERMAL HYDRAULICS, CHEMISTRY, TMI-2 DECAY POWER AND FISSION PRODUCTS, CONTAINMENT EFFECTIVENESS, RADIATION RELEASES, ALTERNATIVE EVENT SEQUENCES, TMI-2 SITE MANAGEMENT, OPERATING PERSONNEL, CONTROL ROOM DESIGN, PROCEDURES, SIMULATORS, EQUIPMENT CONSERVATISM, SAFETY DESIGN MARGINS, PORV DESIGN AND PERFORMANCE, CONDENSATE POLISHING SYSTEM, QA, SECURITY, PAST ACCIDENTS AT NUCLEAR FACILITIES, COST OF THE ACCIDENT, IODINE FILTER PERFORMANCE, AND WASH 1400.
EYWORDS INCIDENT;THREE MILE ISLAND 2 (PWR);REACTOR, PWR;THERMAL HYDRAULIC ANALYSIS;CONTAINMENT;CONTROL PANEL/ROOM;FISSION PRODUCT, IODINE;SECURITY;OPERATING EXPERIENCE;FISSION PRODUCT RELEASE

870/0000001-00000767/ 62
CCESSION NO. 00E0162365
ITLE RELATIVE RELIABILITY ANALYSIS OF THE DIGITAL REACTOR SHUTDOWN SYSTEM ALTERNATE ARCHITECTURES
UTHOR(S) BOWERS TL
ORPAUTH WESTINGHOUSE ELECTRIC CORP., MADISON, PA.
ATE 1980
YPE R
EMO WARD-SR-94000-7 +. 110 PPS, FIGS, REFS, APRIL 1980
VAIL AVAILABILITY - LIMITATIONS ON DISTRIBUTION; SEND REQUESTS TO DOE TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830
ATEGORY 090000;230000
DITION 0121
ORP CODE WEB
OUNTRY A
BSTRACT THIS STUDY EVALUATED THE RELIABILITY OF SIX DIFFERENT ARRANGEMENTS OF DIGITAL ARCHITECTURES FOR THE REACTOR SHUTDOWN SYSTEM TO BE DEVELOPED FOR USE IN A LARGE BREEDER REACTOR PLANT. THE PURPOSE OF THE STUDY WAS TO PROVIDE THIS RELATIVE RELIABILITY INFORMATION WHICH COULD BE COMBINED WITH RESULTS FROM SUBSEQUENT EVALUATIONS OF FUNCTIONAL AND PERFORMANCE REQUIREMENTS ON THESE ARCHITECTURES TO ULTIMATELY SELECT THE SYSTEM TO BE DEVELOPED. (EWH)
EYWORDS WESTINGHOUSE;RELIABILITY ANALYSIS;PROBABILITY;FAULT TREE ANALYSIS;INSTRUMENT, DIGITAL;COMPUTER CONTROL;SHUTDOWN SYSTEM, SECONDARY;REACTOR, BREEDER;REACTOR, LMFBR

870/0000001-00000767/ 63
CCESSION NO. 00X0162362
ITLE PRACTICAL METHODS FOR ESTIMATING THE CONFIDENCE LEVEL UPPER BOUND OF A BINOMIAL LAW ACCORDING TO THE NUMBER OF OBSERVED FAILURES (INCLUDING THE ZERO FAILURE CASE) (IN FRENCH)
UTHOR(S) SIGNORET JP;GEORGIN JP
ORPAUTH CEA DEPARTEMENT DE SURETE NUCLEAIRE, FRANCE
ATE 1980
YPE R
EMO DSN 353 + FRRSR-271 +. 178 PPS, FIGS, REFS, MARCH 1980
ANGUAGE OTHER LANG
VAIL AVAILABILITY - SUSAN DISILVESTRE, DOCUMENT MANAGEMENT BRANCH, DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL, U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, D.C.
ATEGORY 010000;230000
DITION 0121
OUNTRY F
BSTRACT THIS REPORT IS A PRACTICAL WORKING TOOL FOR ALL THE RELIABILITY ENGINEERS TRYING TO EVALUATE EITHER A FAILURE PROBABILITY AS A FUNCTION OF EXPERIMENTS RESULTS, OR A NUMBER OF EXPERIMENTS TO PERFORM AS A FUNCTION OF A PROBABILITY FIGURE TO PROVE. (EWH)
EYWORDS PROBABILITY;ANALYTICAL TECHNIQUE;FAILURE MODE ANALYSIS; THEORETICAL INVESTIGATION;RELIABILITY ANALYSIS;FOREIGN EXCHANGE; FRANCE

870/0000001-00000767/ 64
CCESSION NO. 00R0162350
ITLE NUCLEAR SAFETY AND RELIABILITY ENGINEERING: SELF-ACTUATED SHUTDOWN SYSTEM DEVELOPMENT ANNUAL PROGRESS REPORT FOR THE PERIOD ENDING SEPTEMBER 30, 1979

R(S)	TOPPER RG;CEEPER MH;SWENSON CE
AUTH	WESTINGHOUSE ELECTRIC CORP., MADISON, PA.
	1980
	G
	WARD-SR-94000-4 +. 93 PPS, FIGS, REFS, MARCH 1980
IL	AVAILABILITY - LIMITATIONS ON DISTRIBUTION; SEND REQUESTS TO DOE TECHNICAL INFORMATION CENTER, P.O. BOX 62, OAK RIDGE, TENN. 37830
ATEGORY	090000;230000
DITION	0121
ORP CODE	WEB
OUNTRY	A
BSTRACT	A DESIGN FOR A SELF-ACTUATED SECONDARY SHUTDOWN SYSTEM WAS DEVELOPED WHICH CONSISTS OF A ROD TYPE ABSORBER ASSEMBLY SUPPORTED BY A TEMPERATURE-SENSITIVE ELECTROMAGNET. BY SELECTING A MATERIAL WITH A CURIE TEMPERATURE COMPATIBLE WITH THE FUEL OPERATING TEMPERATURES, THE SUPPORT SYSTEM HAS BEEN DESIGNED SO THAT A REACTOR OVERPOWER OR OVER TEMPERATURE TRANSIENT WILL CAUSE THE MATERIAL TO BECOME PARAMAGNETIC AND TO RELEASE THE ABSORBER ASSEMBLY. SUBSEQUENT EFFORTS WERE DIRECTED AT DEMONSTRATING PROOF OF PRINCIPAL BY SUCCESSFUL TESTS IN ARGON AND SODIUM. (EWH)
EYWORDS	REVIEW;WESTINGHOUSE;RELIABILITY, SYSTEM;REACTOR SHUTDOWN; SHUTDOWN SYSTEM, SECONDARY;EQUIPMENT DEVELOPMENT;R AND D PROGRAM;REACTOR, BREEDER;REACTOR, LMFBR;RELIABILITY ANALYSIS

870/0000001-00000767/ 65

CCESSION NO.	00J0162345
ITLE	DETERMINISTIC CRITERIA VERSUS PROBABILISTIC ANALYSES: EXAMINING THE SINGLE FAILURE AND SEPARATION CRITERIA
UTHOR(S)	WEAVER WW
ORPAUTH	BABCOCK & WILCOX, LYNCHBURG, VA.
ATE	1980
YPE	O
EMO	10 PPS, 2 TABS, 3 FIGS, 34 REFS, NUCLEAR TECHNOLOGY, VOL. 47, PP. 234-43 (FEB. 1980)
ATEGORY	090000;230000;180000
DITION	0121
ORP CODE	MAL
OUNTRY	A
ABB	NUAT
BSTRACT	THE PRIMARY PURPOSE OF DESIGN REQUIREMENT CRITERIA FOR REACTOR SAFETY SYSTEMS IS TO BETTER ENSURE THE ADEQUACY OF SAFETY SYSTEM DESIGN, THE SINGLE FAILURE AND SEPARATION CRITERIA BEING CASES IN POINT. HOWEVER, STRICT ADHERENCE TO THESE TWO CRITERIA, FOR EXAMPLE, MAY ACTUALLY RESULT IN A LESS THAN OPTIMAL DESIGN IN TERMS OF SYSTEM RELIABILITY. WORKING WITHIN THE SPIRIT OF THESE CRITERIA, AN INTEGRATION OF PROBABILISTIC ANALYSES INTO THE LICENSING REVIEW PROCESS FOR SAFETY SYSTEM DESIGN WOULD RESULT IN A MORE RELIABLE SYSTEM, WHICH, AFTER ALL, IS THE INTENT OF DETERMINISTIC CRITERIA. CURRENT PROBABILISTIC ANALYSIS TECHNIQUES ARE ADEQUATE FOR EVEN COMPLEX SAFETY SYSTEM DESIGNS. (EWH)
EYWORDS	REACTOR PROTECTION SYSTEM;DESIGN CRITERIA;PROBABILITY; EXAMINATION;SINGLE FAILURE CRITERION;INDEPENDENCE;RELIABILITY ANALYSIS;SAFETY ANALYSIS

870/0000001-00000767/ 66

CCESSION NO.	00E0160793
ITLE	NUCLEAR POWER PLANT FIRE PROTECTION - PHILOSOPHY AND ANALYSIS
UTHOR(S)	BERRY DL
ORPAUTH	SANDIA NATIONAL LABS., ALBUQUERQUE, N.M.
ATE	1980
YPE	R
EMO	SAND80-0334 +. 70 PPS, 7 TABS, 10 FIGS, MAY 1980
AIL	AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
ATEGORY	120000;230000
DITION	0118
ORP CODE	AUA
OUNTRY	A
BSTRACT	THIS REPORT COMBINES A FIRE SEVERITY ANALYSIS TECHNIQUE WITH A FAULT TREE METHODOLOGY FOR ASSESSING THE IMPORTANCE TO NUCLEAR POWER PLANT SAFETY OF CERTAIN COMBINATIONS OF COMPONENTS AND

SYSTEMS. CHARACTERISTICS UNIQUE TO FIRE, SUCH AS PROPAGATION INDUCED BY THE FAILURE OF BARRIERS, HAVE BEEN INCORPORATED INTO THE METHODOLOGY. BY APPLYING THE RESULTING FIRE ANALYSIS TECHNIQUE TO ACTUAL CONDITIONS FOUND IN A REPRESENTATIVE NUCLEAR POWER PLANT, IT IS FOUND THAT SOME SAFETY AND NONSAFETY AREAS ARE BOTH HIGHLY VULNERABLE TO FIRE SPREAD AND IMPORTANT TO OVERALL SAFETY, WHILE OTHER AREAS PROVE TO BE OF MARGINAL IMPORTANCE. SUGGESTIONS ARE MADE FOR FURTHER EXPERIMENTAL AND ANALYTICAL WORK TO SUPPLEMENT THE FIRE ANALYSIS METHOD.

EYWORDS FIRE PROTECTION;FIRE;POWER PLANT, NUCLEAR;FAULT TREE ANALYSIS

870/0000001-00000767/   67

CCESSION NO. 00E0160792
ITLE A METHOD FOR THE ESTIMATION OF THE RESIDUAL ERROR IN THE SALP APPROACH FOR FAULT TREE ANALYSIS
UTHOR(S) CONTINI S;ASTOLFI M
ORPAUTH JOINT RESEARCH CENTRE, ISPRA ESTABLISHMENT, ITALY
ATE 1980
YPE R
EMO EUR 6750 +. 56 PPS, REFS, 1980
VAIL AVAILABILITY - EUROPEAN COMMUNITY INFORMATION SERVICE, 2100 M ST., N.W., SUITE 707, WASHINGTON, D.C. 20027
ATEGORY 090000;230000
DITION 0118
OUNTRY I
BSTRACT MANY ANALYTICAL ALGORITHMS AND COMPUTER PROGRAMS HAVE BEEN DEVELOPED FOR THE ANALYSIS OF FAULT TREES WHICH ARE BASED ON TWO DIFFERENT APPROACHES, THE "A POSTERIORI" AND THE "A PRIORI". IN THE LATTER ONLY THE MOST SIGNIFICANT MINIMAL CUT SETS (MCS) ARE SEARCHED FOR AND THE OTHERS ARE NEGLECTED. THE ONLY OBJECTION MADE TO THIS APPROACH IS THAT THE USER CANNOT KNOW THE OVERALL PROBABILITY OF THE SET OF NEGLECTED MCS. THIS REPORT ILLUSTRATES THE ALGORITHMS IMPLEMENTED IN THE SALP-MP CODE FOR THE ESTIMATION OF THIS PROBABILITY OR RESIDUAL ERROR.
EYWORDS FAULT TREE ANALYSIS;ERROR ANALYSIS;COMPUTER PROGRAM;ITALY; COMPUTER PROGRAM

870/0000001-00000767/   68

CCESSION NO. 00J0160775
ITLE UNAVAILABILITY OF COMPONENTS WITH INSPECTION AND REPAIR
UTHOR(S) VAURIO JK
ATE 1979
YPE J
EMO 16 PPS, 1 FIG, 6 REFS. NUCLEAR ENGINEERING & DESIGN, 54(3), PP. 309-24 (NOV. 1979)
ATEGORY 110000;010000;170000;230000
DITION 0118
OUNTRY A
ABB NEDE
BSTRACT A GENERAL SET OF INTEGRAL EQUATIONS AND SOLUTIONS HAVE BEEN OBTAINED FOR THE UNAVAILABILITY OF STANDBY COMPONENTS. THE MODEL USES GENERAL FAILURE TIME, TEST DURATION AND REPAIR TIME DISTRIBUTIONS. RANDOM FAILURES AS WELL AS CONTRIBUTIONS FROM HUMAN TESTING ERRORS AND TRUE DEMANDS HAVE BEEN TAKEN INTO ACCOUNT. OPTIMAL TESTING/INSPECTION INTERVALS HAVE BEEN OBTAINED FOR PRACTICAL INSPECTION POLICIES. (FAH)
EYWORDS MAINTENANCE AND REPAIR;AVAILABILITY;EQUIPMENT;ENGINEERED SAFETY FEATURE;ANALYTICAL MODEL;INSPECTION;FAILURE;INCIDENT, HUMAN ERROR;TEST INTERVAL;RELIABILITY ANALYSIS

870/0000001-00000767/   69

CESSION NO. 00R0160730
ITLE LOAD COMBINATION PROGRAM. PROGRESS REPORT NO. 5
THOR(S) CHOU CK;GILMAN FM;DUTTON JC
RPAUTH LAWRENCE LIVERMORE LAB., CALIF.
TE 1980
PE G
MO NUREG/CR-1624 + UCID-18674 +. 56 PPS, FIGS, SEPT. 1980
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 180000;110000;230000
ITION 0118
RP CODE L2C
UNTRY A

RACT   OVERALL OBJECTIVES ARE: DEVELOP A METHODOLOGY FOR APPROPRIATE COMBINATION OF DYNAMIC LOADS FOR NUCLEAR POWER PLANTS UNDER NORMAL PLANT OPERATION, TRANSIENTS, ACCIDENTS, AND NATURAL HAZARDS. ESTABLISH DESIGN CRITERIA, LOAD FACTORS, AND COMPONENT SERVICE LEVELS FOR APPROPRIATE COMBINATIONS OF DYNAMIC LOADS OR RESPONSES. DETERMINE THE RELIABILITY OF TYPICAL PIPING SYSTEMS, BOTH INSIDE AND OUTSIDE THE CONTAINMENT STRUCTURE, AND PROVIDE THE NRC WITH A SOUND TECHNICAL BASIS FOR DEFINING THE CRITERIA FOR POSTULATING PIPE BREAKS. DETERMINE THE PROBABILITIES OF A LARGE LOCA INDUCED DIRECTLY AND INDIRECTLY BY A RANGE OF EARTHQUAKES. REPORT GIVES A GENERAL DESCRIPTION OF THE PROGRAM BY PROJECT AND TASKS, TOGETHER WITH FINANCIAL SUMMARIES, TECHNICAL REPORTS GENERATED, AND MEETING ATTENDANCE. (FAH)

EYWORDS   POWER PLANT, NUCLEAR;DESIGN;TRANSIENT;ACCIDENT;RELIABILITY ANALYSIS;PROBABILITY;PIPES AND PIPE FITTINGS;HJCK;NRC-RM

870/0000001-00000767/   70

CCESSION NO.   0030160729
ITLE   PROBABILITY-BASED DESIGN CRITERIA FOR NUCLEAR STRUCTURES
UTHOR(S)   RAVINDRA MK
ORPAUTH   SARGENT & LUNDY, CHICAGO, ILL.
ATE   1980
YPE   U
EMO   8 PPS, 4 TABS, 10 REFS, NUCLEAR ENGINEERING & DESIGN, 59(1), PP. 197-204 (JULY 1980) (FROM 5TH SMIRT CONFERENCE, AUG. 1979)
ATEGORY   110000;230000
DITION   0118
RP CODE   SEE
OUNTRY   A
ABB   NEDE
BSTRACT   PRESENTS A METHODOLOGY FOR DEVELOPING PROBABILISTIC DESIGN CRITERIA FOR NUCLEAR STRUCTURES. THE MAIN FEATURES OF THE METHODOLOGY ARE CALIBRATION, SECOND-MOMENT SAFETY INDEX PROCEDURE, AND USE OF AVAILABLE DATA ON LOAD AND MATERIAL VARIABILITIES. EXAMPLES ARE INCLUDED TO ILLUSTRATE THESE CONCEPTS. (FAH)
EYWORDS   PROBABILITY;DESIGN CRITERIA;POWER PLANT, NUCLEAR;STRUCTURAL INTEGRITY;STEEL;CONCRETE

870/0000001-00000767/   71

CESSION NO.   0030160715
TLE   A PROPOSAL FOR A REFERENCE CLASSIFICATION OF LWR SYSTEMS TO BE USED IN THE EUROPEAN RELIABILITY DATA SYSTEM (ERDS)
THOR(S)   CORETTI M;MELIS M;MANCINI G
RPAUTH   UNIV. OF PISA, ITALY
TE   1980
PE   U
MO   14 PPS, FROM 3RD EUROPEAN RELIABILITY DATA BANK SEMINAR; BRADFORD, U.K., APRIL 9-11, 1980
AIL   AVAILABILITY - COMMISSION OF THE EUROPEAN COMMUNITIES, D.G.XIII-DIRECTORATE A. BATIMENT JEAN MONNET, PLATEAU DU KIRCHBERG, BOITE POSTALE 1907, LUXEMBOURG (MENTION PAPER 19531 DRA)
TEGORY   010000;230000
ITION   0118
UNTRY   X
STRACT   A DESCRIPTION OF THE STRUCTURE AND OF THE CONTENT OF THE CLASSIFICATION IS GIVEN FOLLOWED BY THE ACTUAL CLASSIFICATION LIST AND TWO EXAMPLES SHOWING HOW TO APPLY IT.
YWORDS   REACTOR, LWR;RELIABILITY, SYSTEM;RELIABILITY, COMPONENT;EUROPE;DATA COLLECTION;ITALY

870/0000001-00000767/   72

CESSION NO.   0030160714
TLE   A PROPOSAL FOR A REFERENCE FAMILY GROUPING CODE AND FAILURE CLASSIFICATION FOR LWR COMPONENTS TO BE USED IN ERDS
THOR(S)   LUISI T;MANCINI G;REDLINGER G
RPAUTH   JOINT RESEARCH CENTER, ISPRA ESTABLISHMENT, ITALY
TE   1980
PE   U
MO   22 PPS, FROM 3RD EUROPEAN RELIABILITY DATA BANK SEMINAR; BRADFORD, U.K., APRIL 9-11, 1980
AIL   AVAILABILITY - COMMISSION OF THE EUROPEAN COMMUNITIES,

U.S.XIII-DIRECTORATE A, BATIMENT JEAN MONNET, PLATEAU DU KIRCHBERG, BOITE POSTALE 1907, LUXEMBOURG (MENTION PAPER E 15536 URA)
EGORY 010000;230000
ITION 0118
UNTRY X
STRACT AS PART OF THE FEASIBILITY STUDY FOR THE EUROPEAN RELIABILITY DATA SYSTEM (ERDS) AND MORE SPECIFICALLY FOR THE COMPONENT EVENT DATA BANK (CEDB), ATTENTION HAS BEEN DEVOTED TO THE SETTING UP OF REFERENCE CLASSIFICATIONS OF SAFETY RELATED COMPONENTS FOR LIGHT WATER REACTORS AND THEIR FAILURE CHARACTERIZATION FOR THE COLLECTION OF RAW DATA ON RELIABILITY AND AVAILABILITY COMING FROM NATIONAL DATA SYSTEM.
EYWORDS LUXEMBOURG;ITALY;RELIABILITY, SYSTEM;RELIABILITY, COMPONENT; AVAILABILITY;DATA COLLECTION;REACTOR, LWR;FAILURE, EQUIPMENT; FAILURE, COMPONENT

870/0000001-00007677 73
CCESSION NO. 00E0160652
ITLE THE PROBABILITY OF INTERSYSTEM LOCA: IMPACT DUE TO LEAK TESTING AND OPERATIONAL CHANGES
UTHOR(S) RODIN MP
ORPAUTH U.S. NUCLEAR REGULATORY COMMISSION
ATE 1980
YPE M
EMO NUREG-0677 +. 26 PPS, 2 FIGS, MAY 1980
VAIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
ATEGORY 120000;010000;230000
DITION 0118
ORP CODE NRC
OUNTRY A
BSTRACT WASH-1400 IDENTIFIED POTENTIAL INTERSYSTEM LOCA IN A PWR AS SIGNIFICANT CONTRIBUTOR TO RISK RESULTING FROM CORE MELT. IN THIS SCENARIO, CHECK VALVES FAIL IN INJECTION LINES OF RHR OR LOW PRESSURE INJECTION SYSTEMS, ALLOWING HIGH PRESSURE REACTOR COOLANT TO ENTER LOW PRESSURE PIPING OUTSIDE CONTAINMENT. SUBSEQUENT FAILURE OF THIS LOW PRESSURE PIPING WOULD RESULT IN LOCA OUTSIDE CONTAINMENT AND SUBSEQUENT CORE MELTDOWN. SIMILAR SCENARIOS ARE ALSO POSSIBLE IN BWR. REPORT EVALUATES VARIOUS PRESSURE ISOLATION VALVE CONFIGURATIONS USED IN REACTORS TO DETERMINE PROBABILITY OF INTERSYSTEM LOCA. IT IS SHOWN THAT PERIODIC LEAK TESTING OF VALVES CAN SUBSTANTIALLY REDUCE INTERSYSTEM LOCA PROBABILITY.
YWORDS REACTOR, PWR;REACTOR, BWR;REACTOR, POWER;ACCIDENT, LOSS OF COOLANT;PROBABILITY;CORE MELTDOWN;VALVE, CHECK;TEST, LEAK; SHUTDOWN COOLING SYSTEM

870/0000001-00007677 74
CESSION NO. 00R0160616
TLE CONTAINMENT ATMOSPHERE RESPONSE (CAR) PROGRAM SECOND STATUS REPORT
THOR(S) LANDONI JA
RPAUTH GENERAL ATOMIC CO., SAN DIEGO, CALIF.
TE 1980
PE G
MO GA-A15582 +. 95 PPS, FIGS, MARCH 1980
AIL AVAILABILITY - NATIONAL TECHNICAL INFORMATION SERVICE, U. S. DEPARTMENT OF COMMERCE, SPRINGFIELD, VA. 22161
TEGORY 070000;120000;110000;230000
ITION 0118
RP CODE LAW
UNTRY A
STRACT THE WORK CONCENTRATED ON DEVELOPMENT OF MODELS DESCRIBING CONTAINMENT PHENOMENA DURING CORE HEATUP IN SUPPORT OF PROBABILISTIC RISK ASSESSMENT STUDIES. MODELS WERE COMPLETED FOR FISSION PRODUCT IODINE SORPTION ON COATED SURFACES, DIFFUSIVITY AND RETENTIVITY OF UNTREATED CONCRETE, IODINE INTERACTION WITH CONDENSING STEAM ON THE CONTAINMENT ATMOSPHERE BOUNDARIES, AND THE CLEANUP FILTER SYSTEM. THESE MODELS WERE INCORPORATED INTO A NEW COMPUTER PROGRAM CALLED CARCAS.
YWORDS FISSION PRODUCT RELEASE;ANALYTICAL MODEL;FILTER SYSTEM; CONTAINMENT;FISSION PRODUCT RETENTION;REACTOR, HTGR;IODINE; SORPTION;CONCRETE;STEAM;AEROSOL;COMPUTER PROGRAM

SSION NO. 0020158047
E DEBRIS-BED COOLABILITY LIMITS, RESULT FROM IN-CORE TESTS D-1, D-2, AND D-3
PAUTH U.S. NUCLEAR REGULATORY COMMISSION, DC
E 1979
PE G
MO 7 PGS, MEMORANDUM W/ENC. TO HAROLD R. DENTON FROM SAUL LEVINE, MAY 10, 1979
VAIL AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)
ATEGORY 120000;050000;060000;230000
DITION 0113
ORP CODE NRC
OUNTRY A
BSTRACT SUMMARIZES RESULTS OF 3 IN-CORE EXPERIMENTS ON COOLABILITY LIMITS OF POST-ACCIDENT PARTICULATE LMFBR FUEL DEBRIS IN A SODIUM POOL WORK. WORK SUPPLIES KEY INFORMATION FOR ASSESSING RISK FROM CORE MELT ACCIDENTS IN LMFBR'S. EXPERIMENTS PERFORMED TO MEASURE UNDER HIGH SODIUM SUBCOOLINGS OF LOF AND TOP LMFBR ACCIDENT SCENARIOS, THE SPECIFIC POWER IN THE BED OF FUEL PARTICULATE AT WHICH DRY OUT OF SODIUM COOLANT OCCURS. RESULTS SHOW LOCAL DRY OUT IS NOT TRUE BED COOLABILITY LIMIT. NRC CONSIDERS RESULTS TO BE BEST DATA AVAILABLE AND RECOMMENDS USING DATA AND MODEL FOR EVALUATING LMFBR DEBRIS-BED COOLABILITY UNDER HIGHLY SUBCOOLED CONDITIONS OF LOF AND TOP ACCIDENTS BECAUSE DRY-OUT LIMIT IS CONSERVATIVE WITH RESPECT TO TRUE BED COOLABILITY LIMIT.
EYWORDS REACTOR, LMFBR;ACCIDENT, TRANSIENT OVERPOWER;ACCIDENT, LOSS OF FLOW;EXPERIMENT;MEASUREMENT;COOLING;SODIUM

1 8/07/0000001-00000767/ 76
CCESSION NO. 0020157773
ITLE STANDARDS FOR INDIVIDUAL OCCUPATIONAL EXPOSURES
ORPAUTH U.S. NUCLEAR REGULATORY COMMISSION, DC
ATE 1979
YPE G
EMO 9 PGS, MEMORANDUM W/ATTACH. TO R. A. PURPLE FROM J. KASTNER, DEC. 13, 1979
VAIL AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE -- MINIMUM CHARGE $2.00)
ATEGORY 150000;190000;230000
DITION 0113
ORP CODE NRC
OUNTRY A
BSTRACT REPORT INCLUDES FOLLOWING CATEGORIES: EXPOSURE OF INDIVIDUALS IN RESTRICTED AREAS, PLANNED SPECIAL EXPOSURES AND UNPLANNED OVER EXPOSURES, SPECIAL CATEGORIES DEALING WITH TRANSIENT WORKERS, WOMEN, EMBRYO/FETUS, AND MINORS, MAJOR ISSUES THAT REQUIRE RESOLUTION SUCH AS QUALITY FACTORS, ICRP-26 (ORGAN DOSE, CAPPING LIMITS, RISK), CAREER OCCUPATIONAL DOSE LIMITS, COLLECTIVE DOSE STANDARDS, GRADUATED CONTROLS, IMPLEMENTATION OF OCCUPATIONAL ALARA CONCEPT, DEPARTURE/DIFFERENCES FROM EXISTING PART 20 REQUIREMENTS, AND EXTERNAL INFLUENCES ON THE DEVELOPMENT OF STANDARDS FOR INDIVIDUAL OCCUPATIONAL EXPOSURES.
EYWORDS AGENCY, NRC;RADIATION EXPOSURE;PERSONNEL EXPOSURE, RADIATION; RADIATION SAFETY AND CONTROL;RADIATION EXPOSURE, RECORD KEEPING

APPENDIX G

DATA SOURCES

Many of the Reliability, Availability and Maintainability (RAM) analyses require failure and repair data. This appendix presents a brief discussion of known data bases with information applicable to RAM analyses.

G

APPENDIX G

DATA SOURCES

The quality and value of most RAM analysis depends greatly on the availability of data. The data required needs to be of good quality, collected with RAM analysis in mind, of sufficient quantity to be statistically significant. The best data are those one has collected himself for which he knowns the limitations. Also, ideally one would like to have data on the specific or identical component or system which is being analyzed. However, data collection of sufficient quality and quantity on all components/systems is usually prohibitative expensive. Thus, one must reley on data collected by others and on component/systems which are similar to the one being analyzed and often in different applications. Potential sources for this data are discussed below.

The following are data sources which have information available for use in RAM analyses:

- o Nuclear Plant Reliability Data System (NPRDS)
- o Generating Availability Data System (GADS)
- o Operating Unit Status Report (NUREG-0020)
- o Licensee Event Report (NUREG-0161)
- o United Kingdom Atomic Energy Authority Data Program (UKAEA) National Center of Systems Reliability (SYREL)
- o IEEE Nuclear Reliability Data Manual (IEEE STD 500-1977)
- o Government-Industry Data Exchange Program (GIDEP)
- o Nonelectric Parts Reliability Data (NPRD-1)
- o Electronic Component Reliability Data (RAC)
- o MIL-HDBK-217D
- o Equipment Manufacturers
- o DoD Information Analysis Centers

A brief description of each of these data sources with the type of information available is presented.

Nuclear Plant Reliability Data System (NPRDS)

The Nuclear Plant Reliability Data System (NPRDS) program is a source of operating reliability statistics for safety-related components and systems in commercially operated U.S. nuclear reactor power plants. The program is operated by the ANS 58.20 Subcommittee to collect engineering, operating, and failure data from electric utilities on a quarterly basis and to report component and system performance data on quarterly and yearly basis.

The scope of reportable systems and components to NPRDS is classified as Safety Class 1 and Safety Class 2 in ANSI Standard N18.2 for pressurized water reactors and in ANS 52.1-1978 for boiling water reactors. Also included in equipment designated as Electrical Class 1E in IEEE STD 380-1975. Information is collected on twenty-nine major categories of components of mechanical and electro-mechanical designs.

Participants in NPRDS are provided with access to (1) complete engineering data on components and system, (2) unit and system operating hours, (3) statistics on reliability performance of equipment, and (4) complete description of component failure including mode, type, cause, effect, and detection.

This information is suitable for design, operations and plant-betterment engineers. The information may be used for reliability and maintainability prediction and assessment and for design-improvement programs.

For additional information, contact:

NPRD System Coordinator
Southwest Research Institute
6220 Culebra Road
San Antonio, Texas 78284
Telephone: 512/684-5111

Generating Availability Data System (GADS)

The Generating Availability Data System (GADS) is a source of summary unit performance data on all types of electric power generating equipment. The GADS program, formerly the EEI Equipment Availability Data System, is operated by the National Electric Reliability Council (NERC) to collect and make available unit performance, pedigree, and event reporting data on nuclear, fossil, hydro, combustion, and combined cycle units.

Information is reported to NERC-GADS by participating utilities on a quarterly basis. Included in GADS are unit statistics, outage event types, outage causes, and unit performance information.

GADS data are suitable for operation and maintenance engineers working mostly with overall plant and major equipment hardware. These data may be used for availability analysis at the unit or component level.

For additional information, contact:

National Electric Reliability Council
Research Park, Terhune Road
Princeton, New Jersey 08540
Telephone: 609/924-6050

Operating Unit Status Report (NUREG-0020)

The Operating Unit Status Report (NUREG-0020) is a source of unit operating statistics for licensed U.S. Nuclear Power Plants. The program is operated by the U.S. Nuclear Regulatory Commission (NRC) to collect daily power levels and monthly operating statistics from utilities and to report (GREY book) statistics across all units on a monthly basis.

Utilities report to NRC monthly on unit performance. Included in the performance statistics are daily average power levels (MWe), operating status (including critical hours, shutdown and online hours, service availability, capacity factors, and forced outage rate), and unit shutdown or power reduction information. Unit shutdown and power reduction records include date, type of reduction (forced, scheduled), duration (hours), reason (equipment, maintenance, refueling, regulatory restrictions, etc.) and major system or component category.

This information is suitable for operating engineers and plant-betterment engineers. The information may be used for availability and unit performance assessment and for plant betterment.

For additional information, contact:

National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151
Telephone: 703/321-8543

Licensee Event Report (NUREG-0161)

The Licensee Event Report (LER) is a source of off-normal and cause and event descriptions of departures from technical specifications in the operation of nuclear power reactor plants. The LER program is operated by the U.S. Nuclear Regulatory Commission (NRC) to collect and make available facility, system, component, and manufacturers' data related to the reported event.

Information is reported to NRC by utilities on a daily, bi-weekly, or monthly basis depending on the nature of the event. Included are details of plant, system, and component statistics, and narrative descriptions as to cause and event.

LER data are suitable for operations and plant-betterment engineers working with safety-related systems and components. These data may be used for sfety analysis and event-sequence studies of major safety systems and components.

For additional information, contact:

License Operation Evaluation Branch
Office of Management and Program Analysis
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Telephone: 301/492-7724

United Kingdom Atomic Energy Authority Data Program (UKAEA)
National Center of Systems Reliability (SYREL)

The United Kingdom Atomic Energy Authority (UKAEA) data program is a comprehesive source of nuclear power reactor reliability data from operating plants within the United Kingdom. The program uses a data classification coding format similar to that of the FARADA and GIDEP programs.

The UKAEA uses its long-standing equipment fault and incident reporting system as a primary source of equipment service data in its reactor plants. There are approximately 900 categories of components in the reporting system. The National Center of System Reliability (SYREL) data bank contains information on performance availability and generic component reliability data.

This information is suitable for maintenance and plant-betterment engineers and may be used for maintenance planning or unit availability studies.

For additional information, contact:

Systems Reliability Service (Date)
UKAEA (SYREL)
Wigshaw Lane
Culcheth, Warrington, Lancashire, WA34NE
United Kingdom

IEEE Nuclear Reliability Data Manual (IEEE STD 500-1977)
(IEEE Guide to the Collection and Presentation of Electrical, Electronic, and Sensing Component Reliability Data for Nuclear-Power Generating Stations)

The IEEE Nuclear Reliability Data Manual is a source of reliability data for electrical, electronic, and electromechanized components commonly is use in nuclear power generating stations. The manual was prepared by Working Group SC5.3 of Subcommittee 5 Reliability of the Nuclear Power Engineering Committee of the IEEE Power Engineering Society. The data contained in the manual were provided by over 200 consultants and experts in the field.

The reliability data presented include failure modes, failure rate ranges, and environmental factor information on generic components. Reliability data appear in the form of hourly and cyclic failure rates and failure mode information for over 1000 electrical, electronic, and sensing components. Low, recommended, high, and maximum failure rates are given for each individual mode.

This information is suitable for desgin engineers. The Data Manual offers a data base to be used for the performance of qualitative and quantitative systematic reliability analyses of nuclear power generating stations.

For additional information, contact:

IEEE Standards Sales
IEEE Service Center
445 Hoes Lane
Piscataway, NJ 08854
IEEE STD 500-1977, SH06684

Government-Industry Data Exchange Program (GIDEP)

Government-Industry Data Exchange Program (GIDEP) is a cooperative program between Government and Industry for exchanging data among participants to reduce time and money for researching relevant areas. The program provides a means of exhanging certain types of technical data used in research, design development, production, and operation of systems and equipment used mainly in electronic or electro-mechanical application.

GIDEP incorporates the Failure Rate Data Program (FARADA), which is jointly sponsored by the Army, Navy, Air Force, and NASA. The FARADA program comprises the collection, analysis, compilation, and distribution of failure rate and failure mode data.

Participants in GIDEP are provided with access to four major data interchanges: (1) Engineerging Data, (2) Metrology Data, (3) Reliability-Maintainability Data, and (4) Failure Experience Data.

GIDEP information is suitable for design engineers working mostly with electronic, electrical, and electromechanical components. This information may be used for qualitative studies such as failure modes and effect analysis, decision tree analysis, and event tree analysis, as well as for quantitative studies such as reliability prediction, test interval calculation, or spare parts studies.

For additional information, contact:

GIDEP Operations Center
Corona, California 91720
Telephone: 714/736-4677

Nonelectric Parts Reliability Data (NPRD-1)

The Nonelectric Parts Reliability Data is a publication providing test and reliability data primarily from military and space applications. The information is provided in four sections: (1) generic level failure rate data, (2) detailed part failure rate data, (3) part data from commercial applications, and (4) failure modes and mechanism.

The nonelectric parts reliability data is suitable for design engineers working mostly with electrical and electromechanical components. This information may be used for qualitative studies such as failure modes and effects analysis and decision tree analysis, as well as quantitative studies such as a reliability prediction of systems composed of nonelectric parts.

For additional information, contact:

Reliability Analysis Center
RADC/RBRAC
Griffiss AFB, New York 13441
Telephone: 315/330-4151

Electronic Component Reliability Data

The Reliability Analysis Center publishes test and reliability data on electronic components, such as integrated circuits, memories, hybrid, linear interface devices and transistors and diodes. The data have been collected from military and industrial sources, analyzed and merged into a common database. The database is used as the source for the publications which include failure rate data, and failure modes and mechanisms information by device type and environmental stress. Summaries are provided which include failure rate comparison by device function, complexity, screening and models in MIL-HDBK-217D (see below).

The electronic components reliability data are suitable for use by design engineers working with electronic systems. The information is very useful for doing reliability prediction, FMEA and FTA on electronic systems.

For additional information, contact:

Reliability Analysis Center
RADC/RBRAC
Griffiss AFB, New York 13441
Telephone: 315/330-4151

MIL-HDBK-217D

This military handbook entitled, "Reliability Prediction of Electronic Equipment," dated 15 January 1982, has been developed to establish uniform methods for predicting the reliability of military electronic equipment and systems. It provides a common basis for reliability predictions and a basis for comparing and evaluating reliability predictions of related or competing designs.

MIL-STD-217D presents failure rate models for nearly all electronic components along with qualifying factors which design or reliability engineers can use to perform reliability analysis on electronic systems in specific applications.

Copies are available from

Naval Publications & Forms Center
5801 Tabor Avenue
Philadelphia, PA 19120
Telephone: 215/697-2000

Equipment Manufacturers

Manufacturers of components, equipment and systems normally have performed some reliability and maintainability testing on their own equipment. Often they will make these data available to the users either routinely or by special request. Great care must be taken when using manufacturers' reliability data since it indicates the quality of their product and they would not release it if it shows poor results.

Manufacturer's data can often be obtained by indicating in the equipment specification that reliability data must be provided.

Manufacturers' data can often provide a second source of information on an component/equipment type but ideally other data should be used.

DoD Information Analysis Center

The DoD sponsors 20 information analysis centers which collect, review analyze appraise summarize and store available technical information on subjects of highly specialized technical nature. In addition to the Reliability Analysis Center which specializes in electronic reliability data, as discussed above, centers have data which are applicable to the design and improvement of nuclear power generating facilities. A few of centers with applicable data are:

- Concrete Technology Information Analysis Center
- Data & Analysis Center for Software
- Hydroulic Engineering Information Analysis Center
- Metals and Ceramics Information Center
- Mechanical Properties Data Center

- Mechanical Properties Data Center
- Nondestructive Testing Information Analysis Center
- Plastics Technical Evaluation Center
- Shock and Vibration Information Center
- Thermal physical and Electronic Properties Information Analysis Center

Information on the availability of data can be obtain by contacting Mr. James F. Pendergast at the Defense Technical Information Center. His address is:

Administrator
Defense Technical Information Center
ATTN: DTIC-AI
Cameron Station
Alexandria, VA 22314
Telephone: 202/274-6260

APPENDIX H

SURVEY QUESTIONNAIRE ON

RELIABILITY PRACTICES AND CONTROL ELEMENTS

This questionnaire was used as a guide to the interviewers during surveys of electric power generating facilities.

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DRAFT 12/22/81
REVISED 2/82

SURVEY QUESTIONNAIRE
ON
RELIABILITY PRACTICES AND CONTROL ELEMENTS

CONTENTS

NOTE: This questionnaire is designed to aid in determining the scope and effectiveness of current reliability practices and control elements planned and applied by the nuclear power generating industry. Although this questionnaire is intended for use with any organization in the industry, it is designed to facilitate gathering of reliability information from the standpoint of those organizations directly responsible for the acquisition and operation of systems and equipment used in nuclear power plants. It is recognized that collecting information on system/equipment acquisition and operation may require an addendum to address specific aspects as they relate to individual organizations. This general questionnaire should be used with addendum, as necessary, in the performance of the surveys.

SURVEY QUESTIONNAIRE

DATE ____________________

ORGANIZATION CONTACTED

NAME

LOCATION

DESCRIPTION

INDIVIDUAL(s) CONTACTED		SURVEY TEAM	
NAME	TITLE/POSITION	NAME	TITLE/POSITION

I ORGANIZATION INFORMATION

1) Is an operating formal reliability organization maintained with a permanent in-house staff?

2) Who is the person responsible for reliability?

What is his position in the organization?

What are the names of other key people in the organization?

3) What is the relationship of reliability to other organizational functions?

Do they interface directly with:

Project Management?

Design?

Quality Assurance?

Other?
(Obtain organization chart, if available)

4) What is the size of the reliability staff?

Is staff composed of experienced reliability engineers?

Number of Professional? Degrees? Average Experience?

Number of Non Professional?

Are supplementary reliability analysts used?

Are reliability related papers published regularly?

5) Describe the general functions of the reliability organization?

- Specification?
  (Prepare, review from reliability standpoint)
- Supplier reliability audit & surveillance?
- Reliability (availability) analysis?
- Design drawing and specification review (for adequacy of reliability requirements)?
- Component engineering?
- Design review (IDR)?
- Demonstration/acceptance test monitoring?
- Failure analysis?
- Data recording & analysis?
- Operational reliability assessment?
- Product improvement programs?

6) Are other organizations involved in reliability (or related) functions?

Test laboratories?

Quality assurance?

Failure analysis?

7) Are formal reliability training programs conducted?

- Reliability engineering training?
- Operational & maintenance training?

II PRACTICES AND CONTROL ELEMENTS DURING ACQUISITION

1) What is the basic approach to assuring system/equipment reliability?

   What reliability indices are employed?

   Are there formal policies/procedures covering:

   - Determination of reliability requirements?

     Are requirements derived from system risk analysis? and optimized with respect to cost?

   - Acquisition of new (or replacement) system/equipment to meet the requirements?

   - Operation and maintenance to assure that reliability is maintained?

2) Are reliability requirements applied contractually on system designers (A&E Firms) and hardware suppliers?

   - Design?

     MTBF | Service LIfe | Forced Outage Rate

     MTTR | Availability | Planned Outage Rate

     Start-up Reliability

     Component Quality?

     Redundance or back-up modes of operation? Fail safe design?

     Ease of maintenance? Diagnostics? Modularity?

   - Test?

     Development/R growth? Screening?

     Demonstration? Acceptance?

     Qualification? 100% Sampling?

   - Management and control?

     R program plan? Failure analysis and reporting?

     Parts control? Data reporting, analyses and feedba

     Critical item control? Reliability assessment?

     Configuration control? Sub-contractor/supplier control?

     Program/design review?

3) Are standard reliability program requirements imposed? (Either fully contractually, limited or as a guide)

MIL STD 785?

MIL HDBK 217?

MIL STD 781?

4) Are system reliability program elements planned and applied?

A. Reliability prediction/apportionment/assessment?

- Allocation techniques?
- Numerical evaluation/prediction analysis?
- Redundancy aspects of passive components?
- Verification test methods?
- SFP detection methods?

The survey should address the objectives of the methods, e.g., are they used to: (1) establish sub-system/component reliability goals, (2) provide quantitative measures of reliability during the development process (3) determine where the design can be improved, (4) aid design trade-off decisions, (5) provide criteria for planning (and verifying) reliability growth and demonstration tests, (6) establish the need for redundancy, (7) identify single failure points and provide quantitative input for early spare provision plans. The survey should obtain information on the extent and effectiveness of system models depicting the reliability interconnection of the subsystems and components, the failure rates (and their uncertainties) used to support reliability predictions; the techniques used to apportion reliability requirements and to set design goals for subsystems and components; and the actual test/field data used to assess achieved reliability.

B. Maintainability prediction/assessment?

- Allocation techniques?
- Numerical evaluation/prediction?
- Verification methods?

The survey should address the objectives of the methods, e.g., are they used to: (1) establish subsystem/component maintainability goals, (2) provide a quantitative measure of how easily a design can be maintained (3) determine where the design can be improved. The survey should obtain information on the effectiveness of the methods to aid design trade-off decisions, plan and verify test methods and provide input for early spare provisioning plans. Maintenance level diagrams, work factors, repair time data (e.g., determined via maintenance analysis and which accounts for human factors and maintenance errors), repair frequencies (e.g., based on component failure rates) and used to support maintainability predictions should be obtained as well as information on techniques to apportion maintainability requirements and set maintainability goals among subsystems and components.

C. System interface?

- Compatability?
- Malfunction effects analysis?
- Detection/evaluation methods?
- Rededial action?
- Consequence analysis?
- FMEA/FMECA?
- Sneak circuit analysis?

The survey should address the effectiveness of analysis techniques in evaluating system interfaces and compatibility between subsystems/components particularly to determine the consequences of failure or malfunction on overall system reliability. Information on how the methods are applied to determine, for example, the need for redundancy and fail safe design features, identify single failure points, identify critical items and to assure subsystem/component compatability should be obtained. Also the survey should address the effectiveness of the methods in providing input to R&M models/predictions, identifying remedial action priorities, identifying critical items, defining failure detection/evaluation methods and providing key inputs for developing maintenance strategies and plans. The extent, depth and rigor of the techniques and in particular the uncertainties of their results should be discussed including: (1) FMEA/FMECA procedures and the basic data and information (e.g., from design configurations, component engineering and part failure rates resulting from prediction studies) used to support the process; (2) structural models (logical "and" and "or" symbols and failure events) used to support fault tree analysis and (3) sneak path analysis techniques that are applied to further locate and ultimately force out potential malfunctions, that occur without component failure, due to the existence of a sneak circuit or latent path.

D. Trade-off study?

- Improvement techniques?
- Cost-effectiveness?
- Improvement evaluation?

The survey should address techniques used to help make R&M trade-off decisions involving the evaluation of design alternatives as well as the determination of program/test requirements. Information on how trade-off studies are performed to determine, for example, the optimum MTBF/MTTR mix that would maximize availability should be obtained including sensitivity curves and other data which would show the relationship of R&M parameters, controls and engineering tasks to availability and cost.

E. Independent Design Review (IDR)?

- Techniques applied?
- Reliability impact?
- Remedial action?
- SFP assessment?
- Techniques to surface hidden system faults?

The survey should address procedures applied to systematically review performance, reliability, maintainability and various other system characteristics at major design and testing decision points. The survey should obtain information on the adequacy and completeness of checklists developed to support design reviews. Criteria applied to determining conformance or adequacy should be obtained covering such design/program items as:

- Program plans
- R&M allocation, predictions and assessments
- Identification and evaluation of critical components
- Test plans and procedures
- Maintenance concepts
- Subsystem and component specifications
- Remedial actions
- Single point failure (SPF) assessment
- FMEA/FMECA/FTA/sneak circuit analysis or other techniques to surface hidden system faults
- Failure analysis reports
- Growth test data
- Production reliability assurance plans
- Supplier control methods
- Configuration management
- Documentation and reports

Also the methods employed to control the independent design reviews including the thoroughness of deficiency follow-up control procedures should be discussed.

5) How do data uncertaintities effect the system reliability program elements?

6) How are the effects of operator and maintenance actions taken into account on safety system reliability analysis?

7) Are <u>hardware</u> reliability program elements planned and applied?

A. Hardware specification?

- Reliability requirements (MTBF, MTTR)?
- Application techniques?

The survey should address techniques used to determine hardware R&M specifications and, in general, how requirements are established that satisfy safety requirements, operational availability needs and also that are attainabl within the state-of-the-art. Information on how quantitative requirements are established and how the requirements are formulated into a hardware specification that reflects an effective balance of the various demands should be obtained.

B. Hardware selection?

- Reliability experience?
- Failure rate considerations?
- Interface considerations?
- Hardware maintainability considerations?

The survey should address basic procedures and criteria applied to select system hardware (e.g., based on proven R&M and long life characteristics and demonstrated acceptability to meet system needs). Information on how failure rate and mode experience data, subsystem/component interfaces (particularly between R&M parameters and the system design and development process), logistic factors and the supplier's background or prior experience in the R&M and relatec areas are considered in selecting critical hardware items should be obtained.

C. Component derating?

- Policy?
- Techniques?
- Are guidelines applied?

The survey should address the use of derating guidelines in the design of hardware items to assure that all components are operated well within recommended stress limits. How techniques are applied to reduce the probability of hardware-induced failures and allow the components to realize the full extent of their inherent reliability should be discussed.

D. Screening?

- Burn-in techniques?
- Testing criteria?
- Selection (part) approval?

The survey should address methods techniques and guidelines used to plan and implement hardware screening and burn-in programs. Information on the application of stress screening during hardware production on a 100% basis for the purpose of revealing inherent, as well as workmanship and process-induced, defects without weakening or destroying the hardware, should be obtained including screen test profiles, time durations, acceptance criteria and other elements and controls. The methodologies and techniques to plan optimum screen programs and to determine the most effective burn-in time periods should be discussed.

E. Production degradation control?

- Method of control?
- Preventive action?
- Acceptance?
- Failure reporting analysis and corrective action?

The survey should address techniques used to control reliability during manufacturing, to minimize degradation of intrinsic or designed-in reliability and to accelerate reliability growth. Information on methods of control, including techniques to isolate intrinsic and induced defects in a manner such that special inspections or screens can be applied to eliminate the defects should be obtained. Information on failure analysis and data collection programs covering failures reported during manufacturing and actual experience during operation and how the data is applied to modify and improve the manufacturing process should also be obtained.

8) Describe any other reliability practices and control elements that are applied during acquisition?

9) Are IEEE and ANS standards effective in producing component reliability?

10) How is component reliability preserved in storage, and during installation and construction?

III PRACTICES AND CONTROL ELEMENTS DURING OPERATION

1) How is reliability assured during operation and maintenance?

Is there an operating philosophy? (e.g., minimum number of hours per start)?

How is operational staff organized?

How is maintenance staff organized?

2) How is operational/failure data collected and analyzed?

Is the data system computerized?

What computer codes are used?

For what?

3) Are operational failures reported, analyzed and fed back to system designers and manufacturers.

4) Is available operational reliability data sufficient?

If not, how can it be improved?

At what cost?

5) Are operational reliability (and availability) assessments performed periodically and reports prepared and issued?

How do data uncertainties effect the reliability assessments?

6) Have product improvements programs been initiated?

7) What reliability program elements are planned and applied?

A. Maintenance policy/practices and strategy?

- Preventive maintenance?
- Corrective maintenance?
- Fault detection/isolation?
- Logistics?
- Downtime control?
- Service (life) time?

The survey should address basic maintenance concepts, maintenance personnel skill levels, support equipment requirements, logistics, training repair management, maintenance manuals and support data and other maintenance parameters. Information on the rationale/cost benefits of the maintenance parameters and techniques for establishing preventive maintenance frequencies, corrective maintenance procedures, fault detection/isolation methods and, in general, controlling downtime over the entire service life of the system/equipment (40 years) should be obtained.

B. Replacement strategy?

- Time constraints? (frequency)
- Replacement criteria?
- Verification?

The survey should address various replacement strategies including those based on time (or number of cycles) constraints as well as those based on the operational condition of the hardware. Information on methods and ratior ale for establishing replacement criteria (time/condition factors), throwaway concepts, and verification that replacement was accomplished properly, and that the hardware is restored to full operational integrity should be obtained.

C. Reliability growth program?

- Failure analysis procedures?
- Reliability improvement techniques?
- Diagnostic activity?
- Requirements verification (MTBF/MTTR)?
- Immaturity failures versus random failures analysis?
- Detection of latent defects during test, method?
- Wearout failure problem, solution?

The survey should address methods applied to analyze, correct, improve and, in general, grow reliability. Information on automatic monitoring functior designed to survey selected system performance parameters or operating conditior (such as temperature) in order to detect impending system/component malfunction and to make (or allow) compensating adjustments or corrections should be obtaine

The extent and depth of built-in hardware diagnostics, the application of end-to-end verification testing, and the rigor and thoroughness of failure/data analysis procedures should be discussed. Also failure/data analysis procedures and activities should be discussed with respect to determining the extent and effectiveness of: (1) analysis techniques to determine root causes as they relate to various hardware technologies; (2) statistical techniques to isolate infant mortality, random and wearout failures and to establish trends, and; (3) control methods to define personnel responsibilities, scheduling requirement: depth of analysis activities, reporting forms, feedback mechanisms and output requirements particularly relative to assessing achieved R&M parameters such as mean time between failure (MTBF) and mean time to repair (MTTR).

IV PROGRAM EFFECTIVENESS

1) During system/equipment acquisition were all of the original program requirements (system/hardware elements) completed in their entirety? R

2) As the programsprogressed, did the attention to the reliability requirements increase, decrease, or stay the same?

3) Which requirements or program elements are considered most cost effective in detecting and correcting failures prior to plant operation?

4) Did significant management changes or organizational changes occur during the programsaffecting the manufacturer, system designer or the utility?

   If so, did this change the attitude regarding the reliability requirements?

5) Were there major changes in the course of the programs such as program stretch-outs, performance definition changes, etc.?

Did the program changes affect the reliability requirements?

6) Did major engineering/design changes occur as a result of reliability deficiencies uncovered during:

- Design (R prediction, FMEA, etc.)?
- Development/R growth tests?
- Demonstration tests?
- Screening?
- Acceptance?
- Plant operation?

7) Were there major problem areas uncovered during the design reviews?

If so, were these resolved satisfactory and in a timely manner?

8) Did significant cost overruns occur during acquisition? To what were these attributed?

Were reliability deficiencies significant contributors to any cost overruns?

9) What reliability research is needed?

10) How much money (and how many manhours) can usefully be spent on reliability?

11) Of what value to reliability are NRC I and E bulletins - which - why?

12) What organization has the best reliability program - why?

13) What NRC sponsored research has been of value?

V RELIABILITY PERFORMANCE INFORMATION

1) In general are the system/equipment used in nuclear power plants considered:

   - Reliable (performs without problems)?

   - Satisfactory (performs in spite of minor problems, requires maintenance, but easy to maintain)?

   - Poor (performs but fails often and requires extensive maintenance)?

   - Unsatisfactory (fails often and requires extensive and difficult maintenance)?

2) What are the principal reliability problems?

   Design?
   Workmanship?

   Operational software procedures?

   Human factors?

   Maintenance procedures?

   Envirnomental considerations?

3) Does actual operational reliability generally agree with what was predicted and measured during acquisition?

4) What subsystems, equipment or components fail most often?

Mechanical?	Outage Rate?
Electrical?	Outage Rate?

DOCUMENTATION

Try to obtain available documentation, such as:

1) Management policy statements concerning reliability.

2) Reliability, maintainability program plans - general and specific.

3) Reliability indoctrination/training programs for management, engineers reliability and QA, operators and maintenance personnel.

4) Procedures or description of methods employed in implementing reliability oriented activities during engineering, testing, production, installation and operation of systems and components.

5) Methods and procedures for system designers and hardware supplier reliability control.

6) Documentation of data feedback system in effect within the organization and any employed universally within the industry.

7) Testing methods and procedures - reliability tests, demonstration tests, acceptance tests, etc.

8) Technical reports of reliability analysis, testing, operations, research or other study efforts; reliability data compilation; maintainability data compilations, life cycle cost analyses.

9) Maintenance procedures, maintenance logs and forms, problem areas, etc. relative to system/equipment operation and maintenance.

10) Warranty practices - typical or standardized warranty policy.

11) Standard or accepted reliability terms and conditions.

12) Published reliability related papers.