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BENCHMARK DESCRIPTION OF CURRENT REGULATORY REQUIREMENTS AND PRACTICES IN NUCLEAR SAFETY AND RELIABILITY ASSURANCE

by

S. L. Halverson, W. A. Bezella,
I. Charak, and C. J. Mueller

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ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

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S. L. Halverson,* W. A. Bezella,
I. Charak, and C. J. Mueller

Reactor Analysis and Safety Division

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*Electronics Division

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ABSTRACT

The objectives of this work are to evaluate and benchmark the current safety and reliability assurance-related practices employed by the NRC. This effort represents an initial phase of a program whose overall purpose is to develop a reliability program (RP). A review of NRC regulations relevant to reliability assurance was made for a boiling water reactor using two representative safety systems; the reactor protection system, and the residual heat removal system. The primary sources of information were the NRC Standard Review Plan and Title 10 of the Code of Federal Regulations, especially Part 50. In addition, relevant regulatory guides, NRC branch technical positions and industry consensus standard were identified and catalogued for the two reference safety systems over the plant's life cycle. The identified standards and criteria were then organized into a RP element matrix of current regulatory requirements organized by life cycle phase, top level assurance function, and items directly auditable by the NRC. A brief review of the licensing process was also undertaken to indicate the effectiveness of NRC implementation of a RP. The results of this work showed that within the NRC regulations a framework already exists in which to integrate, not add, a reliability assurance program.

NRC FIN No.	Title
A2242	Reliability Assurance Program

Table of Contents

List of Tables

List of Figures

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NOMENCLATURE

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ACRS	Advisory Committee on Reactor Safeguards
AE	Architect Engineer
AEOD	Office of Analysis and Evaluation of Operational Data
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASB	Auxiliary Systems Branch
ASLB	Atomic Safety Licensing Board
ASME	American Society of Mechanical Engineers
BF1	Brown Ferry Reactor Unit 1
B&PV	Boiler and Pressure Vessel
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CFR	Code of Federal Regulations
CMF	Common Mode Failure
CP	Construction Permit
CPSER	Construction Permit Safety Evaluation Report
CSB	Containment Systems Branch
DE	Division of Engineering
DFS	Draft Environmental Statement
DEQA	Division of Engineering and Quality Assurance
DFO	Division of Facility Operations
DOD	Department of Defense
DOL	Division of Licensing
DRA	Division of Risk Analysis
DSI	Division of Systems Integration
DST	Division of Safety Technology
EAB	Events Analysis Branch
EEI	Edison Electric Institute
EPM	Environmental Project Manager
EQB	Equipment Qualification Branch
ER	Environment Report
ESF	Engineered Safety Features
FAA	Federal Aviation Administration
FES	Final Environmental Statement
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
FTA	Fault Tree Analysis
GDC	General Design Criteria
HFB	Human Factors and Safeguards Branch
HFEB	Human Factors Engineering Branch
HFS	Division of Human Factors Safety
ICSB	Instrumentation and Control Systems Branch
IE	Office of Inspection and Enforcement
IE/PSB	Program Support Branch
IEEE	Institute of Electrical and Electronics Engineers
IITRI	Illinois Institute of Technology Research Institute
INPO	Institute for Nuclear Power Operation
ISA	Instrument Society of America
LC	Life Cycle
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection System
LPM	Licensing Project Manager
LQB	Licensee Qualifications Branch
LWA	Limited Work Authorization
LWR	Light Water Reactor
MEB	Mechanical Engineering Branch
MOV	Motor Operated Valve
NASA	National Aeronautics and Space Agency
NPRDS	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OL	Operating License
OLB	Operator Licensing Branch
ORPM	Operating Reactor Project Manager
PDR	Public Document Room
PRA	Probabilistic Risk Assessment
PRB	Division of Reactor Programs
PSAR	Preliminary Safety Analysis Report

PSB	Power Systems Branch
PTB	Program Technology Branch
PTRB	Procedures and Test Review Branch
QA	Quality Assurance
QAB	Quality Assurance Branch
RA	Reliability Assurance
RAB	Regulatory Analysis Branch
RAM	Reliability Availability Maintaina
RAP	Reliability Assurance Program
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Resea
RG	Regulatory Guide
RHR	Residual Heat Removal
RO	Reactor Operator
ROAB	Reactor Operations Analysis Branch
RPS	Reactor Protection Systems
RRAB	Reliability and Risk Assessment Br
RRB	Reactor Risk Branch
RSB	Reactor Systems Branch
RSCB	Research and Standards Coordinatio
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SRO	Senior Reactor Operator
SRP	Standard Review Plan
TLR	Top Level Requirements
TV	Tennessee Valley Authority
TMI	Three Mile Island

Project Overview

The work described herein is part of the Reliability Program research project sponsored by the Division of Risk Analysis, Office of Research, NRC. The overall purpose of this research project is to develop and recommend a Program of coordinated reliability engineering and management techniques (i.e., elements) that could interface with on-going industry and regulatory programs to help licensees achieve and maintain an acceptable level of nuclear power plant safety over the lifetime of the plant. The Reliability Program would complement a quality assurance program by establishing the needed reliability levels for structures, systems, components, and operations including procedural and personnel actions important to safety; the QA program would then assure that established procedures to attain these levels are followed.

Prompted by the TMI accident, NRC, DOE, and EPRI have all sponsored studies of aerospace, commercial aircraft, and military programs to understand their approach in optimizing the safety, reliability, and costs of key systems in these programs. NRC staff involved in the Indian Point Hearings, the Salem incident reviews, and ATWS rulemaking have all recommended Reliability Programs with varying descriptions as a potential means for cost-effectively maintaining LWR safety. Recent NRC-sponsored research has identified Reliability Program elements practiced in these other industries as having potential for use in the nuclear industry. Studies to date have made only rather broad generalizations on the benefits or costs of such programs or their activities. The Reliability Program research project discussed here will use these previous recommendations and study results to develop a viable Program and its associated activities, subject this program to detailed evaluation through regulatory analysis and in plant testing, and recommend a final Program that meets the objectives described above.

The research performed to date has been as follows:

(1) The work in this report, viz. benchmarking the existing regulations and their implementation relevant to reliability assurance and organizing them by life cycle phase, top level reliability assurance function, and material directly auditable by the NRC. This will allow a basis for comparison with reliability related regulations and their implementation from other safety-critical, high technology programs such as exist in the aerospace, military, and airline industries. It also provides an information base to facilitate future decision-making with respect to integrating reliability-related regulations with existing regulations. It has shown that within the current body of NRC rules, requirements and guidance, a framework already exists in which to integrate, not add, a reliability-based regulatory program. The most obvious missing ingredients are numerical performance standards tied to risk and reliability and standards outlining the degree of detail that reliability studies should have. Regulations on failure reporting and corrective actions are not currently tied to a Reliability Program framework.

(2) Determining the risk-dominant attributes of the residual heat removal and reactor trip systems for the Browns Ferry Unit-1 plant from a review of existing PRA information for that plant, related LERs, and plant safety

literature including emergency procedures. The purpose was to identify the parameters that govern the unavailabilities of specific, but representative, safety-related systems for nuclear plants so as to indicate the most important aspects of a reliability program. This work is documented in a NUREG/CR to be released in mid-1984. It confirmed, not surprisingly except perhaps for degree, the dominance of dependent (common cause) failures on risk-important sequences involving complex nuclear systems and highlighted the importance of the operator(s) being able to recover safety functions during an accident.

(3) The development of a preliminary Reliability Program structure and associated activities for the operating phase of a nuclear power plant. This development provides a baseline for subsequent detailed evaluation and value-impact analysis. Previous recommendations of earlier industry and government groups and consultants were factored into this structure. Synopses of current licensee practices and regulatory requirements that address the same safety issues addressed by the Reliability Program were provided. Relevant new industry initiatives or regulatory activities were cited. Finally, preliminary qualitative judgements on potential value-impact were made. This work is documented in a NUREG/CR to be released in mid-1984. The key elements of the Reliability Program model developed for the operation phase are a systems reliability analysis program; a parallel plant/systems performance monitoring program; subprograms that perform the continuous integration of these with Operations and Maintenance requirements and activities; and a distinct subprogram to deal with accident recovery issues.

(4) Support work, as yet unpublished through NUREGs, surveying the most useful related activities in the aerospace, military, and airline industries. This was done by focusing on specific programs or practices, namely NASA's Space Shuttle Main Engine Program, the Navy's Trident Missile Guidance System Program, FAA certification of airframes, and FAA operation of ground control systems. This work was performed by Charles Stark Draper Laboratory (NASA, Navy, and FAA certification) and Reliability Technology Associates (FAA operations) under contract to ANL. The purpose was to identify the reliability activities that could best be integrated with existing plant practices and NRC regulations to enhance overall safety. Significant conclusions were centered on FAA certification and related that certain FAA practices have attractive features which might be applicable to the nuclear industry. These include the use of industry representatives who monitor and approve various production and manufacturing phases; the maintenance practices of the FAA in many aspects, e.g. certification of personnel, FAA/industry interactions, failure experience feedback, and reliability-centered-maintenance; and the anonymous reporting of the "Aviation Safety Reporting System".

On-going and future work will develop a Reliability Program structure and associated activities for the complete lifecycle of a power plant. Regulatory analysis and in-plant demonstration programs will be used to define the content of and tailor these elements so that they are responsive to what licensees feel is most warranted. Of course, the Reliability Program must also meet the NRC's purpose in helping to assure that operating safety is maintained at an adequate level through plant life. Regulatory value-impact analysis will also be used in conjunction with the demonstration program to identify the most cost-effective ways of implementing a Reliability Program, both from an industry and from a regulatory standpoint.

C. J. Mueller
ANL Principal Investigator

Executive Summary

This work benchmarks the current safety and reliability assurance-related requirements employed by the NRC on two risk-important systems, the reactor protection system (RPS) and the residual heat removal system (RHR) of a nuclear power plant. Current regulatory requirements are condensed into a matrix identifying the corresponding auditable elements of a reliability program (RP). The purpose is to establish a basis for comparative evaluation with RPs from other high technologies. This evaluation would indicate the current regulatory requirements and industry practices that could be integrated with or replaced by elements of a formal reliability program more relevant to safety and reliability, more conducive to the licensing process, and more easily auditable by the NRC in the operational phase of a plant.

The task was initiated with a comprehensive review of the NRC Standard Review Plan (SRP, NUREG-0800) and Title 10 of the Code of Federal Regulations (10 CFR), especially Part 50 (10 CFR 50). These were the primary but not sole sources of information for the requirements analysis. The SRP is essentially a compendium of acceptable solutions to meeting the requirements in 10 CFR. By checking all references in the appropriate SRP chapters for the reference (RPS and RHR) systems, the regulatory guides, NRC branch technical positions, and industry consensus standards having implications for a RP were identified and catalogued. The completed catalogue (Table 3.1 this report) of RP-relevant standards and criteria describes where the standards/criteria are invoked, the major interfaces with supporting documents, and their applicability over each life cycle phase of a nuclear plant.

Using as a model the reliability assurance programmatic elements recommended by IITRI and the Rome Air Development Center in a previous NRC study, the standards and criteria identified above were then organized into these elements. The resulting matrix of current regulatory requirements organized by life cycle phase, top level reliability assurance function, and material directly auditable by the NRC, provides the requirements benchmark for comparison with the RPs from the FAA, NASA, and DOD. It also provides the baseline for integrating or replacing current requirements with an RP-oriented licensing approach. Unless elimination of some current requirements can be guaranteed, an RP will serve as a mechanism for ratcheting additional requirements. The results of this work (compiled in Table 3.2 of this report) showed that within the current body of NRC rules, requirements and guidance, a framework already exists in which to integrate, not add, a reliability-based regulatory program.

To benchmark the actual NRC practices in monitoring safety system operations, a summary review of current NRC practices relevant to reliability assurance of the residual heat removal system in the operations phase of the life cycle was included. Reduction of NRC monitoring requirements is another objective of this research program. As reference material for subsequent comparison with the FAA regulatory organization (Chapter 4), a description of the various NRC organizational units that have responsibilities in reliability-related efforts has been included as Appendix A.

Also undertaken as part of this task was a brief review of the licensing process to indicate how and where NRC implementation of a performance-oriented

RP could be used to reduce licensing efforts and schedules. This licensing review with indications of potential RP impacts on licensing is reported in Chapter 2 of the current task report. FAA evolution toward regulating performance rather than the details of achieving performance (e.g., design details) has strongly enhanced regulator-industry cooperation in the aircraft certification (analogous to licensing) process and is judged to have enhanced safety. It is clear that use of a quantitative performance-oriented RP in licensing in conjunction with safety goals could also reduce the plant-to-plant variability in safety posture implied by the spread in current PRA results (see Inside NRC - 1/24/83) by imposing risk-oriented availability requirements on safety-related systems. Finally, a detailed summary of the current (non-reliability-based) licensing process is also provided as Appendix B to allow in-depth comparisons with the FAA approach, the review of which is being undertaken in another part of RP research.

Because of the difficulties in obtaining the necessary information for review of the licensee and vendor practices, the output described above is limited to the current NRC requirements and practices that relate, explicitly or implicitly, to requirements of a RP. However, the framework for factoring in industry practices has been established and it should be a reasonably small effort to include this following efforts. To factor in how and at what cost the industry assures compliance with the reference system regulatory requirements through system tech specs, operating, testing, maintenance, and QA procedures and to identify RP alternatives would require industry cooperation to facilitate meaningful results.

1. Introduction and Summary

The purpose of the work presented in this report is to benchmark the current regulatory approach to assuring the safety and reliability of nuclear systems. This benchmarking will facilitate the comparison of the Nuclear Regulatory Commission (NRC) approach with safety and reliability assurance approaches used in the military, aerospace, and avionics technologies. This comparison will ultimately be used to select reliability assurance elements that comprise a reliability assurance program (RAP) that could serve as an option to current regulatory practices in assuring the safety and reliability of nuclear power plant-. Accordingly, the potential of a RAP to simplify/ improve aspects of the rent regulatory approach will be identified as part of this benchmark description. Thus, the work in this report is an initial step in the development of a RAP for the NRC.

This work is part of an overall research program undertaken by the Argonne National Laboratory for the Division of Risk Analysis of the Office of Research in the NRC. The objective of this research is to develop and demonstrate a pilot reliability assurance program as a regulatory option designed to provide maximum assurance that plant safety levels are maintained. This research is supportive of a trend in which the NRC has been attempting to move the regulatory process away from the details of plant design and operation. A RAP should also provide a formal auditable process by which the NRC can assure public health and safety. As pointed out in the Federal Register, Vol. 46, No. 226 dated 11/24/81, a number of diverse regulatory initiatives are supportive of this trend toward self-regulation by the utilities with NRC auditing. Among them is the proposal of the TMI Action Plan that licensees use probabilistic risk assessment (PRA) methods as design and operations management tools. The use of PRA results to establish system availability requirements that a RAP must achieve and maintain is an obvious step in this direction and the mechanism for integrating these results is part of the RAP research program. Table 1.1 presents the recently proposed quantitative safety goals for nuclear power plants.

Figure 1.1 is an NRC program chart showing major elements involved in ensuring that safety-related systems meet availability requirements so as to maintain overall plant safety. An integration of these elements with the information provided from risk assessments into a pilot RAP and subsequent demonstration of this program on a reference safety system are major aspects of the overall research. By folding in life-cycle costing models already established in other high technologies with system reliability requirements and costs, RAP can be used to assure cost-effective decision-making to the obvious benefit of the industry. RAP will also provide to the NRC documented, auditable results to facilitate the NRC's evaluation that plant safety levels have been maintained. The demonstration is expected to establish that a RAP: (1) is a more direct and relevant approach to assuring safety than the current regulatory approach, (2) is more geared to auditable compliance with safety goals by quantifying risk-related requirements, and (3) provides a means of enhancing plant availability while maintaining safety.

In support of the overall research in developing and demonstrating a RAP, the role of the work documented herein is to provide a benchmark description of current regulatory and industry practices in assuring the safety and reliability of the reference system throughout its lifecycle. The current regulatory requirements that affect the safety and reliablity of the reference safety system are described in terms of their

- Applicability at different phases in the lifecycle (design, pre-op testing, operation, maintenance etc.).
- Applicability in assuring the meeting of top level RAP requirements (design, assurance, system availability assurance, etc.).
- Potential for use in providing information that could be audited by the NRC in monitoring a utility RAP.
- Current industry/NRC interfaces and relevant documentation in estabishing and assuring compliance.

Also summarized is how the NRC monitors the safety aspects of the reference system in the operational phase. (The analogous description for other phases of the lifecycle should be a follow-on study.)

The final step in the completion of this task is to summarize how the industry assures the reference system safety, reliability, and regulatory compliance through the system tech specs, operating, testing, maintenance, and quality assurance procedures and scope the staffing requirements. From the above information and analogous information from RAP approaches in other high technologies it will be possible in later tasks to identify the weaknesses in the current regulatory and industry approaches to system safety and reliability assurance. Based on this information a RAP geared to the needs of the nuclear industry can be formulated and tested.

The rest of this report is organized as follows. Chapter 2 provides an overview of the philosophy of the NRC with respect to safety and reliability and how it enters into the licensing process. Since one of the goals of a RAP is to simplify the licensing process, a brief summary of the current approach to licensing is outlined with an indication of where RAP could be used to simplify the process. Appendix A provides a more detailed description of this process for ultimate comparison with the FAA process to be undertaken in another phase of RAP research.

Chapter 3 discusses the relevance of current regulatory requirements and guidelines in reliability assurance. These guidelines and supporting documentation are organized by their applicability in various phases of the life-cycle of a nuclear plant. Their potential for providing auditable information for the various elements of a RAP as depicted in Figure 1.1 are also highlighted. The purpose is to establish a basis for integrating a RAP option with existing regulatory requirements. For a RAP to serve as a regulatory option that receives the support of the industry, it is clear that it must replace selected regulatory requirements and mesh with others. Otherwise it will be perceived, and rightly so, as an instrument that imposes more requirements on an already overburdened industry. It is shown in this chapter that auditable elements of a RAP already exist within the current requirements, but are scattered throughout these requirements. Thus, the implementation of a RAP would not necessitate new requirements so much as a folding of existing requirements into a more systematic and readily auditable format specified by a formal reliability assurance program.

Chapter 4 describes how the NRC monitors the safety of a typical safety system by assuring its compliance to the technical specifications and operating requirements created to satisfy the current requirements on plant safety. A brief assessment of the staffing requirements required to perform this monitoring is provided. The purpose is to establish a basis for cost-benefit comparisons between the current NRC approach with a RAP auditing approach. A summary description of NRC groups having reliability assurance-oriented responsibilities has been provided to identify potential areas of consolidation in the NRC auditing process.

Chapter 5 summarizes the results of this assessment of the current NRC regulatory practices with regard to reliability. Future efforts are identified to scope how the current approach could be strengthened by its replacement and/or integration with elements of reliability assurance programs from other high technologies. In a sense this section anticipates the results of future RAP tasks. A follow-up, in-depth assessment of this is a necessary requirement to implementing a viable RAP option.

Table 1.1 Proposed* Safety Goals for Nuclear Power Plants

Qualitative Goals

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant accidents such that no individual bears a significant additional risk to life and health.

- Societal risks to life and health from nuclear power plant accidents should be as low as reasonably achievable and should be comparable to or less than the risks of generating electricity by viable competing technologies.

Numerical Guidelines

Prompt Mortality Risk

- The risk to an individual or to the population in the vicinity of a nuclear power plant site of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

Delayed Mortality Risk

- The risk to an individual or to the population in the area near a nuclear power plant site of cancer fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes.

Benefit/Cost

- The benefit of an incremental reduction of risk below the numerical guidelines for societal mortality risks should be compared with the associated costs on the basis of $1,000 per man-rem averted.

Plant Performance

- Large-Scale Core-Melt: The likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000 per year of reactor operation.

*NUREG-0880, February 1982 (for comment)

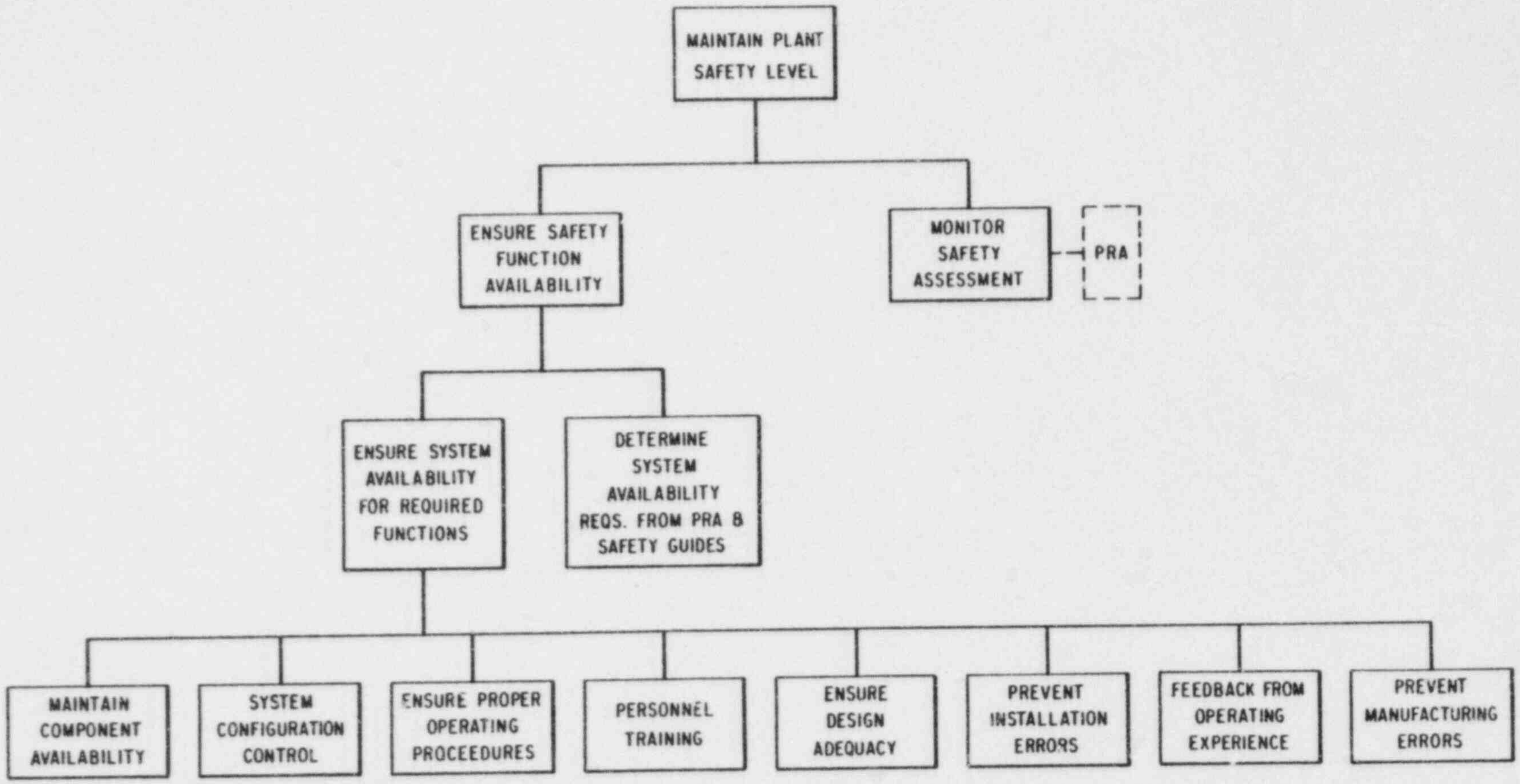

Figure 1.1 Major NRC Availability Requirements Elements

2. RAP Vis-à-vis the Current Licensing Process

The purpose of this chapter is to briefly describe the licensing process to show how and where a reliability assurance program might be used to simplify that process. Although reliability is not mandated through formal regulatory requirements for a reliability assurance program, system safety and reliability are implicitly assured by the licensing process in a "three level of safety" approach. A statement of these levels is: 1) prevent accidents, 2) accommodate anticipated and unlikely faults, and 3) protect the public against extremely unlikely faults. Table 2.1 presents definitions of this qualitative safety approach. Further elaboration is beyond the scope of this report -- what is important to realize is that a RAP approach will facilitate the quantification of probability of success of these levels by formally establishing and then maintaining safety system performance and availabilities requirements. Thus, RAP is supportive of, not contrary to, the current safety philosophy.

The licensing process and the inclusion of safety and reliability elements will be briefly summarized here to establish where a RAP option could be used to simplify and speed up the licensing process. A more complete description of the present regulatory process is given in Appendix A of this report. Figure 2.1 indicates the three-part process used by the NRC to review the application of a utility for a construction permit.

The first part includes the Preliminary Safety Analysis Report (PSAR). The PSAR presents the design criteria and preliminary design information for the proposed reactor as well as comprehensive data on the proposed site. The report also discusses various hypothetical accident situations and the safety features that will be provided either to prevent accidents or, should they occur, to mitigate their effects on both the public and the facility's employees. The PSAR includes sufficient information to allow for a substantive review and evaluation of the applicant's quality assurance (QA) programs that cover design and procurement. Thus, the safety and reliability elements that would be covered in RAP are currently treated in the PSAR, either explicitly or implicitly.

The second part includes the environment and site suitability information that provides a basis for a review of the environmental impact of the proposed power plant. Since the applicant's environmental report and the NRC's environmental impact statement include consideration of accidents, and since risk-based accident evaluations are a major element of RAP, RAP can have a considerable influence on the risk management aspects of the initiation phase and subsequent course of accidents. Hence, a RAP option has the potential to impact the environmental review process where currently largely deterministic safety evaluations are performed.

The third part includes antitrust information for legal review by the appropriate governmental organizations. Obviously, this last part is unrelated to RAP.

When an application is submitted, it is first reviewed to determine whether it contains sufficient information for a detailed review. It is formally accepted (docketed) only if it meets certain minimum acceptance criteria. It is then reviewed to determine that the plant design is consistent with NRC regulations, regulatory guides, and other regulatory positions. Design methods and calculational procedures are examined to establish their validity. Checks of actual calculations and other procedures of design and analysis are made by the NRC staff to establish the validity of the applicant's design and to determine that the applicant has conducted his analysis and evaluation in sufficient depth and breadth to support required findings related to safety. The review checks tend to concentrate on design details rather than process.

The principal features of the staff's review, including the potential impact of RAP on this review process, can be summarized as follows:

1. The population density and use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology and hydrology are reviewed. Siting criteria are specified in 10 CFR 100. Engineered safety features (ESFs) and systems provided for control of radiological effluents from the plant are evaluated. The integration of risk importance in RAP could be used here to quantify the need for and reliability requirements of ESFs, with obvious potential benefits both in safety and economy by placing the more stringent requirements on more risk important systems.

Although straying somewhat from the potential impact of RAP on the licensing process, it is emphasized here that there is a clear potential for economic advantage of a RAP-based design. The NRC-defined quality group classification system is contained in Regulatory Guide 1.26.[a] The arbitrary categorization of systems and components into various quality group classifications has a direct impact on the design, fabrication, and testing of those systems and components, and hence on costs. The assignment within a higher or lower quality group takes no account of a component's importance to risk. Reliability specifications based on risk importance could be used to specify an appropriate quality group on a more rational, defensible basis than is currently employed in Regulatory Guide 1.26.

2. The response of the facility to various anticipated operating transients and to a broad spectrum of hypothetical accidents is evaluated. These hypothetical accidents are then evaluated conservatively to determine that the calculated potential offsite doses would not exceed the guidelines for site acceptability given in 10 CFR 100. Programs of basic research and development necessary to assure the resolution of safety questions, along with a schedule for completion of the work showing that such safety questions will be resolved prior to operation of the facility, are reviewed. As noted above, the specification within RAP of accident analyses concentrating on risk-important scenarios has the potential to focus both analytic and design efforts and reduce unnecessary and costly conservatisms usually associated with the deterministic approach to reactor safety.

3. The facility design and programs for fabrication, construction, and testing of the plant structures, systems, and components important to safety are reviewed. These include programs proposed to verify plant design features and confirm design margins. Plans for the conduct of plant operations including the organizational structure, the technical qualifications of operating and technical support personnel, the measures taken for industrial security, and the planning for emergency action following hypothetical

[a] Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, U. S. Nuclear Regulatory Commission Regulatory Guide 1.26, Revision 2 (June 1976).

accidents that might affect the general public are reviewed. Since a comprehensive RAP covering the complete life cycle from design through operation would be an integral part of these programs and plans, reduction in staff time and effort would occur if an audit of RAP practices could replace or facilitate the current review. Auditable elements of current practice that could be part of a RAP audit are described in Chapter 3.

If the staff concludes that acceptable criteria, preliminary design information and financial information are documented in the application, a Safety Evaluation Report (SER) is prepared by NRC. The SER is a summary of the staff review and evaluation of the application relative to the anticipated effect of the proposed facility on the public health and safety. A RAP-based application with facility design geared to risk importance would force the quantification of a new plants effect on public safety at an early stage. This should increase the staff's confidence that its review has focussed on the proper issues.

As soon as an application for a construction permit is docketed, copies of the PSAR are provided to the Advisory Committee on Reactor Safeguards (ACRS), an independent statutory committee established to provide advice to the NRC on reactor safety. An ACRS subcommittee reviews each application for a construction permit for a commercial nuclear power plant. Normally, before the full committee considers a project, the staff provides its SER for the committee's information. This staff report and the report of the ACRS subcommittee form the basis for committee consideration of a project. The full committee meets at least once with the staff and with the applicant to discuss the application. When the committee has completed its review, its report is submitted via a public letter to the Chairman of the NRC.

An environmental review is also performed by the NRC staff and its consultants to evaluate the potential environmental impact of the proposed plant, as well as to provide comparisons between the benefits to be derived and the possible risk to the environment. After completion of this review, a Draft Environmental Statement (DES) containing conclusions on environmental matters is issued. The DES is circulated for review and comments by the appropriate federal, state and local agencies as well as by individuals and organizations representing the public. After receipt of all comments and resolution of issues, a Final Environmental Statement (FES) is issued.

When the construction of the nuclear facility has progressed to the point where much final design information and plans for operation are ready, the applicant submits the Final Safety Analysis Report (FSAR) in support of an application for an operating license. The FSAR sets forth the pertinent details on the final design of the facility including design of the reactor, containment, engineered safety features, waste handling systems, and all other safety-related structures, systems, and components. The FSAR also supplies plans for operating and procedures for coping with emergencies. The role of a RAP and its integration with design and operation would not change, although its elements would be more clearly defined. Thus, comments on the potential of RAP to strengthen or simplify the review process as discussed above apply here also. The NRC staff again reviews the information and prepares a distinct SER for the operating license. As for the construction permit, the ACRS again makes an independent evaluation and presents its advice to the Commission by public letter.

Each operating license contains Technical Specifications that set forth the particular safety and environmental protection measures required for the facility and the conditions of operation required to assure public health and safety and protection of the surrounding environment. Through its inspection and enforcement program the NRC maintains surveillance over construction and operation of a plant throughout its lifetime to assure compliance with Commission regulations. NRC surveillance relevant to assuring reliability of safety-related systems during operation are treated in Chapter 4. By making the auditing surveillance more responsive to plant performance, akin to the FAA's recognizing and rewarding performance of maintenance programs for aircraft, it would appear that these functions of the NRC could be made more effective.

The above discussion of the benefits of a RAP are necessarily referred to as potential, since the development of a RAP is a major outcome of research yet to be completed. However, the following brief review of certain statistics associated with the present licensing process leads to the conclusion that a carefully implemented RAP approach to licensing could be used to reduce the extreme uncertainties and excessive times associated with the current process.

Construction permit review times of one year or less were common in the late '60s. The current NRC estimate for an uncontested construction permit is 19 months. However for reactors currently under construction, the record for longest elapsed time from docketing to permit issuance is 76 months (Shearon Harris). Another plant of the same type, designer and size docketed at about the same time received its permit after only 21 months (Summer 1). Although this large difference may be at least partly due to non-NRC-related project delays, the inability to predict the duration of NRC reviews with any confidence is the rule not the exception.

Less uncertain is the fact that the time consumed in NRC licensing reviews is large. An estimate of the NRC manpower in the Office of Nuclear Reactor Regulation (NRR) required to review license applications can be derived from information recently published in the Federal Register relevant to NRC license fees.[a] Based on actual NRC costs in FY 1981, the direct person-years of effort related to construction permit reviews ranged from 20 to 29 per case. The comparable range for operating license reviews was 24-28.

At this point, one can only speculate on how much a RAP could shorten the review process and reduce NRC and licensee manpower requirements. However, the fact that RAP would be more systematic than the present process implies that a RAP-based licensing process, perhaps combined with certain elements of the present system, could be beneficial both to licensees and regulators. The experience of the Federal Aviation Administration (FAA), based on our discussions with both aircraft industry and FAA personnel in support of other aspects of RAP research, supports the conclusion that a performance-oriented RAP would serve to stabilize the licensing process and reduce the current adversarial climate between the NRC and the nuclear industry.

[a] Proposed Revision of License Fee Schedules, 47 Federal Register 52454 (1982).

Table 2.1 The Level of Safety Approach

Three Levels of Safety (LOS) Protected Against Design Basis Events

LOS 1 provides reliable plant operation and prevention of accidents during normal operation

LOS 2 provides protection against anticipated and unlikely events by preventing propagation of these events into serious accidents

LOS 3 provides for acceptable plant response to extremely unlikely events

Design Basis Events are Categorized by Frequency and Severity

Normal Operation Events include any steady power operation or departure from operation that is expected to occur regularly or frequently. Occurrence results in no damage or significant loss of effective fuel lifetime.

Anticipated or Upset Events include any off-normal condition which individually may be expected to occur once or more during the plant lifetime. Occurrence results in an operational incident which involves (1) no reduction of effective fuel lifetime below the design values and (2) accommodation with at most a reactor trip that assures the plant will be capable of returning to operation after corrective action to clear the trip cause.

Unlikely or Emergency Events include any off-normal condition which individually is not expected to occur during the plant lifetime; however, when integrated over all components and systems, events in this category may be expected to occur once or more during the life of the plant. Occurrence results in a minor incident which involves a general reduction in the fuel burnup capability and at most a small fraction of fuel rod cladding failure.

Extremely Unlikely or Faulted Events include any off-normal conditions of such low probability that no events in this category are expected to occur during the plant lifetime, but which nevertheless represent extreme or limiting cases of failures. Occurrence results in a major incident which involves substantial fuel and/or cladding melting or distortion in individual fuel rods but with the configuration remaining coolable.

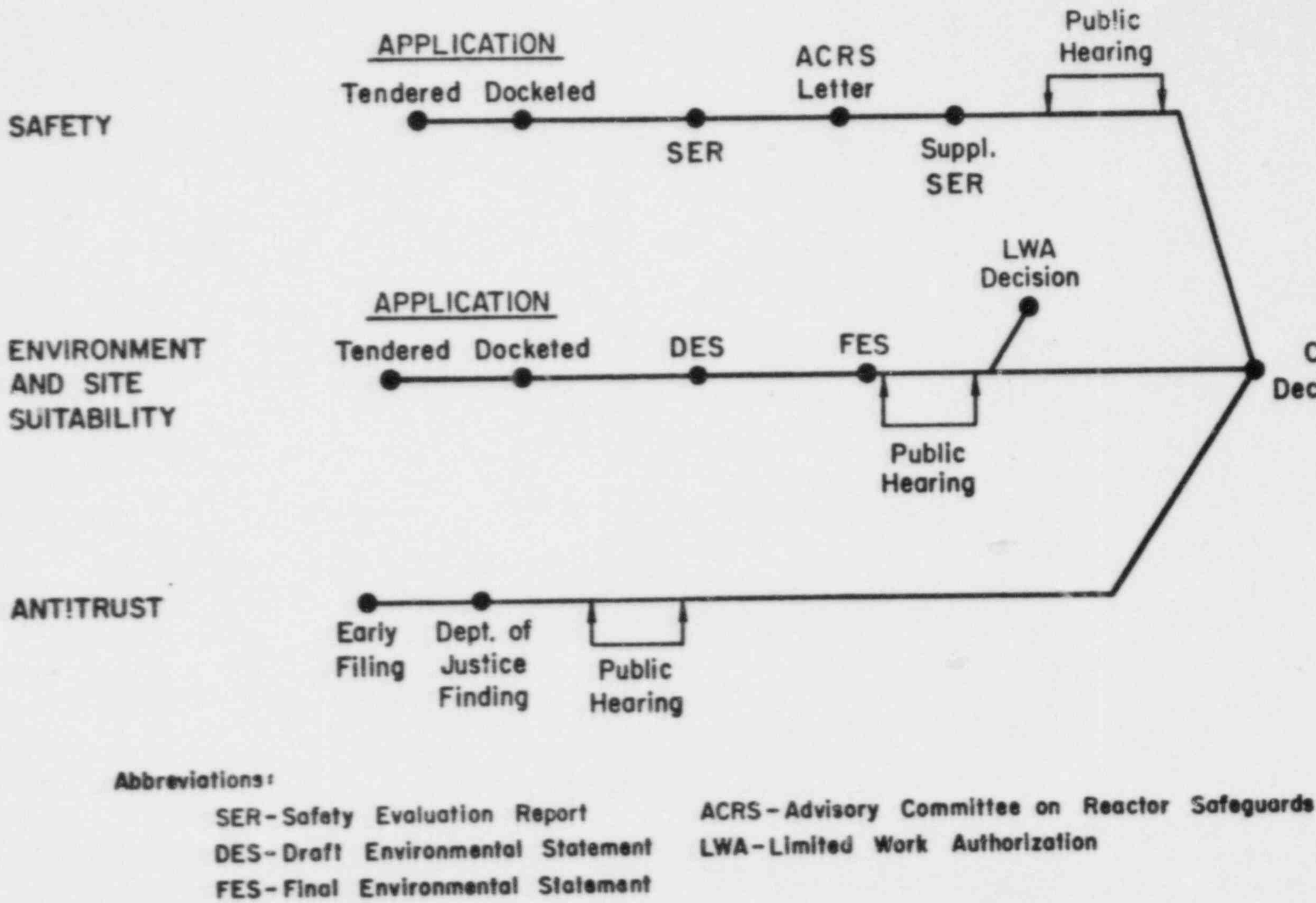

Figure 2.1 Parallel Tracks in Construction Permit Review Process

3. Current Regulatory Requirements and Guidelines Relevant to Reliability Assurance

This chapter reviews current regulatory requirements and guidelines that serve to assure reliability for two selected reference systems, namely the reactor protection system (RPS) and the residual heat removal system (RHR) for a boiling water reactor. The alternative reliability assurance approach(es) that will be investigated as part of the overall research to develop a recommended RAP for the nuclear industry must be evaluated relative to the current (NRC) approach to reliability assurance. This chapter establishes a basis for this comparative evaluation by identifying the documents that define and support the regulatory requirements for the selected systems and organizing them according to the role they have or could have in a RAP approach to assuring systems reliability over the lifecycle of a plant. The reference systems were chosen to focus the needs, requirements, and demonstration studies for a RAP. However, they are representative of the systems used to protect a reactor and as such, serve to provide a more focused basis for evaluating RAP approaches to assuring system safety and reliability.

The review herein has been based on the reliability related standards and criteria embodied in the "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants" (Reference 1), subsequently referred to as the SRP, and Title 10 of the Code of Federal Regulations (10 CFR). The information contained in these documents has been reviewed for its relevance within a RAP and by life cycle as mentioned above.

In addition, background material on the reliability functions within the different offices of NRC has been reviewed and is summarized in Appendix B. Those offices that check compliance with the regulatory requirements are discussed below. The scope of the analysis herein is limited to 10 CFR and the ancillary requirements of the NRC Office of Nuclear Reactor Regulation (NRR), the Office for Analysis and Evaluation of Operational Data (AEOD), and the Office of Inspection and Enforcement (IE). Licensee and manufacturer reliability assurance practices relative to the RPS were unavailable for this review. However, some information on the Tennessee Valley Authority (TVA) Browns Ferry Unit 1 (BF1) RHR including the Safety Analysis Report (SAR), preoperational test results, a draft systems analysis, technical specifications and surveillance requirements were obtained to identify preliminarily the reliability assurance program elements presently incorporated by TVA. Further reviews will allcw a correlation to be made between cause (regulatory requirements) and effect (designer/vendor/licensee reaction) and further identify reliability assurance program elements for the RPS and RHR. Potential difficulties in relating the existing body of regulations to specific TVA reliability program elements will then be identified.

The scope of the RPS review covered all plant life cycle phases as described above. The scope of the RHR study was limited to the operational phase of the life cycle of a nuclear power plant, and discusses the applicable documents in the context of the primary and secondary reliability program elements defined below. This serves to focus on the current implementation of NRC policy related to reliability and safety.

The source documents for NRC/NRR acceptance and guidance criteria that can be related to a reliability assurance program (RAP) are 10 CFR and the SRP. SRP Sections include Sections 3.10 (Seismic Qualification), 3.11 (Environmental Design of Mechanical and Electrical Equipment), 5.4.7 (Residual Heat Removal System), 6.3 (Emergency Core Cooling System), 7.2 (Reactor Protection System), 7.3 (Engineered Safety Features), 7.4 (Safe Shutdown Systems), and applicable sections of Chapters 13 (Conduct of Operations), 14 (Initial Test Program), 17 (Quality Assurance), and 18 (Human Factors). Chapter 15 (Accident Analysis) also contains reliability-related and RPS-specific acceptance criteria and was also included within the scope of the SRP review. The references in each SRP chapter identify the applicable sections of 10 CFR as well as any applicable Regulatory Guides (RGs), Branch Technical Positions (BTPs), and Industry Standards.

Each SRP chapter was reviewed and those documents related to reliability of the RPS or RHR, either explicitly or implicitly, were tabulated in Table 3.1. Each document is identified by an alphanumeric code, title, the SRP chapter in which it is invoked, its authority, primary interfacing or ancillary documents, the life cycle phase or phases of the program to which it applies, and the system to which it applies as follows:

Interfacing Documents: The primary interfaces between the documents listed in Table 3.1 are identified in the interfacing documents column. Since these interfaces are sometimes circuitous, a graphic interface diagram is shown in Figure 3.1 to aid in showing these interrelationships for the BF1 RPS. The BF1 RHR interfaces are similar in nature and therefore have not been presented.

SRP Chapter: The documents which provide the bases for NRC licensing review are identified in the NUREG-0800 column by chapter number and the force of their authority as either acceptance criteria (A) or guidance (G) by a check mark.

Life Cycle Phases: The phases of the life cycle (LC) of the reactor to which the criteria apply are identified as follows:

(1) Conceptual Design
(2) Design/Development/Procurement
(3) System Integration (including construction, installation and preoperational testing)
(4) Operations (including maintenance and periodic testing).

Applicability: The reference system or systems to which the criteria are applicable are identified by a bullet in the appropriate column.

In addition to documents invoked by the NRC in the SRP, Table 3.1 lists supporting documents relevant to the reliability of the Browns Ferry RPS and RHR but not specifically invoked in the SRP. These documents, identified in the interfacing documents column, are included because of their potential future applicability to the RAP and for further consideration in extensions of this study.

The acceptance criteria and guidance provided by the documents listed in Table 3.1 for the reactor protection system can be grouped according to their role in a specific lifecycle (LC) phase(s) of the system. Table 3.2 provides this grouping and further organizes the current requirements for the RPS within a hierarchy of RAP elements. These elements and their functions are described as follows:

TLR Program Elements Each of the LC categories is divided into primary reliability assurance program elements, called top level requirements (TLRs), comprising the following:

(1) Management
(2) Design Assurance
(3) Component/System Availability
(4) Operating Reliability Assurance
(5) Experience Feedback.

All of the NRC regulatory requirements reviewed in this report fall into one or more of these TLR categories.

RAP Sub-elements and Auditables Aggregated under each TLR are the RAP sub-elements, unique to each LC phase, and the auditable items which are provided by the vendor, architect-engineer, or utility in conformance to the regulatory requirements and guidance. The compliance documentation provided by the NRC after an audit is also identified.

Organization Interfaces The table also identifies branches within the NRC Office of Nuclear Reactor Regulation (NRR) with primary and secondary responsibilities for the performance of a review or audit. Additionally, during the operations phase the Office for Analysis and Evaluation of Operational Data (AEOD) and the Office of Inspection and Enforcement (IE) have the primary responsibility for the handling and processing of operating information including Licensee Event Reports (LERs) and non-compliance and discrepancy reports. IE assumes a primary role in Quality Assurance. The interrelationships and cooperative interfaces between the various NRC offices and branches is discussed in detail in Appendix A to this report.

Metrics The "metrics" column identifies the qualitative bases which the NRC use to gauge the acceptability of the specified auditables. Where these metrics are too extensive to list, a blanket reference to the SRP acceptance criteria and guidance or other applicable criteria is given.

The present reliability assurance-related practices of the NRC are those defined by the documents invoked by the SRP as specified in Table 3.1. Although the term "reliability" occurs explicitly in some of the documents, it is usually used in a qualitative sense and in rare cases implies quantitative reliability criteria. None of the invoked documents describes a reliability assurance program with the same degree of detail that 10 CFR 50, Appendix B, and its ancillary standard ANSI N45.2 describe a Quality Assurance Program. However, since a reliability assurance program depends upon the existence of an effective quality assurance program, those documents which are quality oriented are also implicitly related to a reliability program.

Those documents that identify reliability-related acceptance criteria or guidance applicable to the reference systems, either explicitly or implicitly, are summarized below and listed according to their positions in Table 3.1. The individual summaries that follow form the basis for the capsulization of the information in Table 3.2. Those standards and criteria in Table 3.1 that are either not relevant to the RHR or RPS or are otherwise ancillary and invoked in other primary standards and criteria are not discussed below.

<u>1C CFR 50</u>

(1a) <u>Section 50.34</u>

This section is invoked in the SRP for both acceptance and guidance for evaluating reliability-related information contained in the Safety Analysis Reports. Paragraph 50.34(a)(7) provides guidance concerning the description of the quality assurance program in the Preliminary Safety Analysis Report and conformance to the Appendix B Quality Assurance Criteria. Paragraph 50.34(b)(6)(iii) requires plans for preoperational testing and initial operations in the Final Safety Analysis Report (FSAR). QA planning is a reliability program sub-element of the management of a reliability program during the conceptual design phase and preoperational testing is a sub-element of design assurance and operating reliability during the systems integration phase.

Paragraph 50.34(b)(6)(i) requires the identification of the applicant's organizational structure, allocations of responsibilities and authorities, and personnel qualification requirements. Paragraph 50.34(b)(6)(iv) requires plans for conduct of normal operations, including maintenance, surveillance and periodic testing of structures, systems, and components. Paragraph 50.34(b)(8) requires a description and plans for an operator requalification program in accordance with 10 CFR 55, Appendix A. These paragraphs are relevant to management and operations reliability TLRs.

(1b) Section 50.55a(h) (IEEE Standard 279)

This standard, used as a basis for acceptance, requires that the reactor protection system meet the following qualitative reliability and maintainability requirements:

(1) General Functional Reliability
(2) The Single Failure Criterion
(3) Minimum Maintenance
(4) Low Failure Rates
(5) Component Derating
(6) Equipment Qualification
(7) Test and Calibration
(8) System Repair.

No explicit references to analytical techniques to be used to demonstrate conformance to these requirements, such as failure mode and effects analysis (FMEA), common mode/common cause failure analysis (CMF/CCF), single failure analysis, and fault tree analysis (FTA), are given. This deficiency is remedied in an ancillary standard, IEEE Standard 379.

This standard is primarily applicable to the design assurance reliability program sub-element during the design and development phase.

(1c) Section 50.55(e)

This section is invoked for both guidance and acceptance of the experience feedback requirement of the reliability assurance program during the design, development and procurement phase and during the system integration phase. It requires the holder of a construction permit to report to the NRC each deficiency found in design and construction within 24 hours (initial notification) followed by a written report within 30 days. Those deficiencies which are reliability-related could become part of a reliability data base.

(1d) Appendix B

Appendix B is invoked as a basis for acceptance of a quality assurance program and is applicable to all life cycle phases of a nuclear plant in the management and planning, design assurance, and experience feedback reliability elements.

Reliability assurance program descriptions are not presently identified as an integral part of Appendix B, nor are specific reliability assurance program elements discussed. However, these elements are implied and fit within the framework of Appendix B.

(1e) Section 50.40(b)

This section is invoked as a basis for acceptance of the technical qualifications of the operating personnel. These requirements are augmented by ancillary Standard ANSI/ANS 3.1, and Regulatory Guide 1.8. These requirements assure operations reliability during the operations phase.

(1f) Appendix A

This appendix describes the General Design Criteria (GDC) applicable to either the RPS or RHR or both as identified in Table 3.1.

GDC 1 - Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance progam shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

GDC 10 - Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC 13 - Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

GDC 15 - Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

GDC 18 - Inspection and testing of electric power systems. Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

GDC 19 - Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

GDC 21 - Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

This GDC is used as a basis for acceptance of both the RPS and the RHR and requires qualitative reliability and testability. It requires conformance to the following criteria:

(1) Single Failure

(2) Redundancy and Independence

(3) Periodic In-service Testing

(4) Failure Detection.

Conformance to minimum redundancy requirements after removal of a component or channel is not required provided that acceptable reliability can be demonstrated. GDC 21 is supported by ancillaries IEEE Standard 338 and Regulatory Guide 1.118.

GDC 23 - Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

This GDC is used as a basis for acceptance of both the RPS and RHR. Appendix A to Section 7.1 of the SRP requires that an FMEA, in accordance with that requested in Regulatory Guide 1.70, should be performed to demonstrate that all RPS failure modes are acceptable, i.e., that the RPS will fail to a "safe" state.

GDC 24 - Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

This GDC is an ancillary to 10 CFR 50.55a(h) previously discussed.

GDC 29 - Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

This GDC is used as a basis for acceptance of the RPS. Appendix A to Section 7.1 of the SRP requires that the extremely high probability of the protection system meeting functional requirements be demonstrated by conformance to deterministic criteria based on probabilistic reliability assessments performed by the NRC.

GDC 32 - Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

GDC 34 - Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

GDC 35 - Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure.

GDC 36 - Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

GDC 37 - Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

(1g) Section 50.55a(g)

This section is invoked as a basis for acceptance of the RPS and RHR components which are part of the reactor coolant system (RCS) pressure boundary. It requires in-service examinations and tests to verify operational readiness and conformance to the ASME Boiler and Pressure Vessel Code, Section XI. This requirement is relevant to the operations reliability element which includes periodic testing and examination of the piping, valves, and other hydraulic components associated with the RPS and RHR that are part of the RCS pressure boundary.

(1h) <u>Section 50.72</u>

This section and its ancillary Regulatory Guide 1.16, while not invoked by NRR in the SRP, nevertheless provides the AEOD and IE requirement for Licensee Event Reports (LERs) that are the source of operating information on which safety analyses and event sequence data are based. The LER is presently used by the Nuclear Plant Reliability Data System (NPRDS) to infer reliability data. This requirement is relevant to the reliability experience feedback element which includes event reports during the operations phase of the life cycle.

(1i) <u>Section 50.73</u>

This section is the subject of a proposed rule change on the Licensee Event Reporting System which is related to the previously described Section 50.72 and is therefore similarly related to the reliability experience feedback element.

The Directors of AEOD and IE requested internal review of proposed final rules associated with : (1) immediate notification requirements for operating nuclear reactors (10 CFR 50.72); and (2) the Licensee Event Report System (10 CFR 50.73). The interfacing NRC offices are being requested to determine whether the scope and content of these proposed final rules would meet their needs for operational safety information.

Both rules have been published for public comment, and comments have been incorporated into the proposed final rules. In addition, the staff has integrated the requirements of 50.72 and 50.73 in order to assure consistency of reporting and to minimize duplication of effort.

(1j) <u>Section 50.59</u>

This section provides the rules governing changes, tests and experiments to be implemented during the operations phase and which are not described in the safety analysis report. It requires a review of all such changes by IE and the appropriate NRC Regional Office and in particular prohibits their implementation if either changes in technical specifications or unreviewed safety questions are involved.

(1k) Section 50.36

This section provides the rules which govern technical specifications regarding safety systems including the RHR and RPS including limiting safety system settings, surveillance requirements, design features and administrative controls, all applicable to the operations phase.

(1l) Section 50.90

In the event that changes in design, tests, or experiments are planned by a licensee which involve a change in technical specifications or involve an unreviewed safety question this section requires that the licensee apply for an amended license or construction permit so that the change can be fully reviewed by NRC.

10 CFR 55

Operations reliablity is presently assured by requirements for operating personnel training, qualification and periodic requalification. This section is supported by Section 50.34(b)(8) and ancillaries Regulatory Guide 1.8 and ANSI Standard 18.1 discussed elsewhere in this report.

ANSI Standards

(3a-3q) ANSI Standard N45.2 and Ancillaries

These standards are invoked by various Regulatory Guides. They provide guidance for the implementation of a QA/RA program in conformance to the requirements of 10 CFR 50, Appendix B previously discussed.

The ancillary standards which are specifically applicable to the RPS and/or RHR during various life cycle phases are identified in Table 3.1.

(3r) ANSI N18.1 (ANS-3.1)

This standard establishes acceptance criteria for the selection and training of nuclear power plant personnel. It is relevant to both the management and operations reliability program elements during the conceptual design phase and operations life cycle phases. It provides acceptance criteria to be used in evaluation of programs dealing with management personnel reliability indoctrination and training and provides the bases for continuing review and revision of these programs as required during the entire life cycle. More directly it is relevant to operations and maintenance personnel training.

(3s) ANSI 18.7 (ANS-3.2)

This standard provides acceptance criteria for management personnel indoctrination and training and provides the bases for planning and implementing organizational modifications during all phases of the life cycle. This is relevant to the management and operations reliability program elements.

(3w) ANSI/ASME BPV-XI

This standard is invoked in 10 CFR 50, Section 50.55a(g) and provides rules for in-service inspection of components of the RPS and RHR.

Branch Technical Positions

The technical bases for some sections of the SRP are provided in Branch Technical Positions or Appendices which are included in the SRP. These documents typically set forth the solutions and approaches determined to be acceptable in the past by the staff in dealing with a specific safety problem or safety-related design area. These solutions and approaches are codified in this form so that staff reviewers can take uniform and well-understood positions as the same safety problems arise in future cases. Some Branch Technical Positions and Appendices may be converted into Regulatory Guides if it appears that this step would aid the review process. Like Regulatory Guides, the Branch Technical Positions and Appendices represent solutions and approaches that are acceptable to the staff, but they are not required as the only possible solutions and approaches.

(4a) BTP CMEB 9.5-1

This branch technical position provides guidance for the provision of fire protection of safety systems. It is relevant to the design assurance program element during the conceptual and design development phases. It is intended to reduce the probability of common-cause failures caused by fires in safety systems which otherwise meet the single failure criterion of 10 CFR 50, Section 50.55a(h) and its ancillary Regulatory Guide 1.53.

This BTP also provides guidance for determining conformance to 10 CFR 50, GDC 3, "Fire Protection," which is a basis for the acceptance of the fire protection system. A fire hazards analysis which is relevant to the reliability-related common mode failure/common cause failure (CMF/CCF) analysis is requested for systems required to bring the reactor to safe shutdown.

(4b) BTP RSB 5-1

This branch technical position provides guidance in satisfying the isolation, pressure relief, pump protection and test requirements of the GDCs applicable to the RHR.

Regulatory Guides

The regulatory guides provide guidance for determining conformance to acceptance criteria and have implied relevance in the following reliability assurance program elements:

(5a) R.G. 1.8 provides acceptance criteria for the selection and training of engineering and technical support personnel in accordance with ANSI N18.1 and for information concerning general personnel qualification requirements of 10 CFR 50, Section 50.34(b)(6)(i). This is relevant to both the management and operations reliability program elements during the conceptual design and operations phases of the plant life cycle.

(5b) R.G. 1.22 provides guidance for periodic testing of protection system actuation in conformance to the requirements of 10 CFR 50, Appendix A, GDC 21. This is relevant to the operations reliability program element which includes periodic testing during the operations phase of the plant life cycle.

(5c) R.G. 1.26 provides guidance and acceptance criteria to use in the design of the RPS and RHR. It establishes ASME Boiler and Pressure Vessel Code requirements for those parts of the system important to safety that are not covered in 10 CFR 50, Section 50.55a. It is relevant to the design assurance program element during the conceptual and detail design phases of the plant life cycle.

(5d) R.G. 1.28 provides general endorsement of ANSI N45.2 and guidance in its application to satisfy the quality assurance requirements of 10 CFR 50, Appendix B. It is relevant to the management program element during the conceptual and detail design and during systems integration.

(5e) R.G. 1.29 provides acceptance criteria and guidance for identifying those systems which should be designed to withstand the effects of a safe shutdown earthquake (SSE) and be designated as seismic Category I systems. It is relevant to design assurance program element during the conceptual design phase and establishes seismic design criteria.

(5f) R.G. 1.30 provides an endorsement of ANSI N45.2.4 and provides guidance for evaluating conformance to 10 CFR 50, Appendix B, requirements for installation and testing of instrumentation and electric equipment associated with the RPS and RHR during the systems integration phase. It assures reliability in design and provides experience feedback.

(5g) R.G. 1.33 provides an endorsement of ANSI N18.7/ANS-3.2 and defines acceptance requirements and guidance for quality and reliability assurance in accordance with the requirements of 10 CFR 50, Appendix B. It requires procedures for start-up, operation, shutdown, and maintenance for systems important to safety, such as the RHR and RPS. This is relevant to the management and operations reliability program elements during the conceptual design and operations phases.

(5l) R.G. 1.58 provides an endorsement of ANSI N45.2.6 and defines both acceptance criteria and guidance for conforming to the requirements of 10 CFR 50, Appendix B. It establishes criteria for qualifying personnel who perform audit, surveillance, and testing functions during the design development, systems integration, and operations phases of the reliability program. This is relevant to the management reliability program element.

(5m) R.G. 1.64 provides an endorsement of ANSI N45.2.11 and defines both acceptance criteria and guidance for conforming to SRP requirements for configuration control management program planning and for design assurance during the conceptual design and design development phases. Specific additional guidance is given to assure that the designer is independent from those individuals or organizations verifying the design.

(5n) R.G. 1.68 provides acceptance criteria and guidance to determine conformance to 10 CFR 50, Appendix A and Appendix B. It provides design assurance during the systems integration phase by preoperational testing (also testing during shutdown and cooldown under normal plant conditions) to verify that components and systems used for shutdown and cooldown are functionally reliable. This will provide information feedback to the designer and the reliability data base and is therefore relevant to design assurance, operations reliability, and information feedback program elements. R.G 1.68 is supported by its ancillary R.G. 1.68.2 which provides guidance for tests that are intended to demonstrate the capability for remote shutdown.

(5q) R.G. 1.70 The SRP also contains a blanket reference to Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," which requests that Failure Mode and Effects Analysis be provided by the applicant in Safety Analysis Reports (SARs) for both the Reactor Protection System and the Engineered Safety Features System. Regulatory Guide 1.70 further branches into specific guidance on Chapter 17 (quality assurance programs) in WASH 1283, 1284, and 1309 (References 2, 3, and 4).

(5s) R.G. 1.88 endorses ANSI N45.2.9 and provides guidance for conformance to the QA/RA records requirements of 10 CFR 50, Appendix B. It is relevant to the information feedback reliability program element.

(5t) R.G. 1.94 endorses ANSI N45.2.5 and provides guidance relevant to the Safety Class Structures which are structures designed to protect Class 1E equipment from the effects of design basis events. It provides guidance during the systems integration phase by defining acceptable bases for demonstrating conformance to 10 CFR 50, Appendix B, by inspection and test and is relevant to the design assurance reliability program element.

(5u) R.G. 1.105 (Rev. 2) endorses ISA Standard ISA S67.04 and, when approved, will provide guidance to assure that the instrumentation provided to monitor variables during anticipated operational occurrences and the controls provided to keep the system within technical specifications meet specified accuracy and drift requirements. This assures operational reliability by requiring that these criteria be included in the design and development phase, that performance be demonstrated in equipment qualification tests, and that calibration be periodically verified during operation.

(5v) R.G 1.116 endorses ANSI Standard N45.2.8 and provides guidance on implementing quality assurance requirements of 10 CFR 50, Appendix B, for installation, inspection and testing of mechanical equipment associated with the BF1 RPS and RHR. It is applicable during the systems integration and operations life cycle phases and is relevant to the design assurance, operations reliability, and information feedback reliability program elements.

(5w) R.G. 1.123 endorses ANSI Standard N45.2.13 and provides acceptance criteria and guidance for determining conformance to 10 CFR 50, Appendix B. It establishes criteria for evaluating QA/RA management, organization, and supplier/designer audit and surveillance during the design, development and procurement phase.

(5x) R.G. 1.144 endorses ANSI Standard N45.2.12 and provides acceptance criteria and guidance for determining the conformance to 10 CFR 50, Appendix B, requirements for both internal and external audits during all phases of the life cycle of the plant. This is relevant to the audit of reliability-related elements of the QA program.

(5y) R.G. 1.146 endorses ANSI Standard N45.2.23 and provides acceptance criteria and guidance for determining conformance to 10 CFR 50, Appendix B, requirements for the training and qualification of audit personnel.

(5z) R.G. 1.16 provides guidance to IE and AEOD in the evaluation of operating information obtained from the licensee as required by 10 CFR 50, Section 50.36, "Technical Specifications." That section requires that each applicant for a license authorizing operation of a nuclear power plant include in its application a set of proposed technical specifications. These technical specifications, as issued by the NRC, are incorporated into the facility license and are conditions of the license. (See Appendix B of this report for a detailed discussion of technical specifications.) Technical specifications are now included as two appendices to the license: Appendix A technical specifications related to health and safety, and Appendix B technical specifications related to environmental impact. Each of these appendices includes a section on reporting requirements. The reporting program described in this regulatory guide involves the reporting requirements of Appendix A technical specifications only.

(5aa) R.G. 1.118 provides guidance for determining conformance to 10 CFR 50, Section 50.55a(h), Appendix A, GDC 21, and IEEE 338. It assures operations reliability by requiring periodic testing and inspection of the RPS and RHR to confirm operational availability. At the present time there are no SRP requirements for reporting a failure detected during testing that could form the basis for experience feedback. However, the guidance in IEEE 338 is that failure data be collected in a recognized failure data collection bank such as the Edison Electric Institute or the Nuclear Plant Reliability Data System (NPRDS).

(5bb) R.G. 1.53 This regulatory guide provides guidance for determining conformance to the single failure criterion of IEEE Standard 279 and its ancillary IEEE Standard 379. It requires that the failure mode and effects analysis be performed, on both the RPS and RHR logic and actuator systems.

(5cc) R.G. 1.89 provides guidance for evaluating conformance to the requirements of Criterion III, "Design Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. That appendix requires that design control measures provide for verifying the adequacy of a specific design feature by design reviews, by calculational methods, or by suitable qualification testing of a prototype unit under the most adverse conditions. This regulatory guide describes a method acceptable to the Regulatory staff for complying with the Commission's regulations with regard to design verification of Class 1E equipment for service in light-water-cooled and gas-cooled nuclear power plants. This guide and its ancillary IEEE Standard 323 are relevant to the design assurance and management reliability program elements during the conceptual and detail design phases.

(5dd) R.G. 1.100 provides guidance for evaluating conformance to the requirements of Criterion III, "Design Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. That appendix requires, among other things, that design control measures provide for verifying the adequacy of design such as by the performance of a suitable testing program. Where a test program is used to verify the adequacy of a specific design feature, it is required to include suitable qualification testing of a prototype unit under the most adverse design conditions. This regulatory guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with respect to verifying the adequacy of the seismic design of electric equipment for all types of nuclear power plants. This guide and its ancillary IEEE Standard 344 are relevant to the design assurance and management reliability program elements during the conceptual and detail design phases.

(5ee) R.G. 1.149 provides guidance in demonstrating conformance to operator training requirements of 10 CFR 55 and is relevant to operations reliability. The need for improvements in operator training in the areas of response to abnormal and emergency situations was highlighted as a result of the operator errors noted in NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report." Use of the actual plant for training operators to respond to accidents would result in additional challenges to the plant's protective features and is therefore undesirable. Thus, the additional training required to improve operator performance should be performed on simulators. In order to maximize the effectiveness of this training, the simulator must be kept current with changes in the reference plant and lessons learned from operating experience. Recommendations of instructors and operator trainees for improving a simulator should be encouraged. ANSI/ANS 3.5-1981, "Nuclear Power Plant Simulators for Use in Operator Training," in conjunction with this regulatory guide, provides guidance in these areas.

NUREG-0737 is a letter from D. G. Eisenhut, Director of the Division of Licensing, NRR, to licensees of operating power reactors and applicants for operating licenses forwarding post-TMI requirements which have been approved for implementation. Following the accident at Three Mile Island Unit 2, the NRC staff developed the Action Plan, NUREG-0660, to provide a comprehensive and integrated plan to improve safety at power reactors. Specific items from NUREG-0660 have been approved by the Commission for implementation at reactors. In this NRC report, these specific items comprise a single document which includes additional information about schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions. It should be noted that the total set of TMI-related actions have been collected in NUREG-0660, but only those items that the Commission has approved for implementation to date are included in NUREG-0737.

The following items are applicable to the BF1 RHR during the operations phase of the life cycle and are relevant to the management and operations reliability sub-elements:

(11a) Item I.A.1.1 establishes requirements for the qualifications of shift technical advisors and descriptions of long term training programs.

(11b) Item I.A.2.1 establishes requirements for the immediate upgrading of training and qualification of reactor operators and senior reactor operators in the recognition and mitigation of accidents.

(11c) Item I.A.2.3 establishes requirements that instructors in training programs must complete the senior reactor operator's exam.

(11d) Item I.A.3.1 establishes requirements for increasing the examination scope and operator qualification standards for both written examinations and examinations on simulators.

(11e) Item I.B.1.2 establishes requirements for the evaluation of the organization and management of resources for the training and qualification of operators in accident recognition and mitigation.

(11f) Item II.B.4 establishes requirements for the development of training programs for mitigating core damage.

(11g) Item II.E.4.2 establishes requirements for containment isolation dependability through implementation of diversity in design and operator training in the definitions of essential and non-essential systems.

(11h) Item II.K.3.21 establishes requirements for the modification of the LPCI system (emergency mode of the RHR) to permit automatic restart on a loss of water level if an initiation signal is still present to assure adequate core cooling under accident conditions. A design change description is required which is relevant to design assurance during the operations phase of the life cycle.

NUREG-0660 is the NRC Action Plan developed as a result of the TMI-2 accident. Two items related to design assurance are applicable to the RHR as follows:

(12a) Item II.E.3.2 establishes requirements for a reliability study of the RHR.

(12b) Item II.E.3.3 establishes the requirements for an analysis of the shutdown heat removal requirements to demonstrate that the RHR meets functional performance requirements.

Problem Areas

Some of the problems in correlating the reliability-related program elements with the SRP are caused by the way the body of regulations and its ancillaries are invoked and the force of authority which they have, as follows.

Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," calls specifically for failure modes and effects analyses (FMEAs) of the RPS and the Engineered Safety Feature System. The FMEA is a recognized analytical tool for reliability analysis. This is reinforced by SRP Section 7.1, Appendix A. Yet, Regulatory Guide 1.70

is not invoked in the applicable SRP chapters as the basis for acceptance of the RPS or RHR but rather in a blanket statement in the Introduction to the SRP. Further, reliability considerations and information required for each section of the FSAR are not specified. Therefore, if Regulatory Guide 1.70 were withdrawn as a regulatory guide and compliance with it made a Commission rule or regulation, the intent of Regulatory Guide 1.70 and its explicit reliability-related elements would be brought firmly within the scope of an audit.

Additionally, Regulatory Guide 1.70, Chapter 17, provides guidance for quality assurance programs over all life cycle phases by invoking references 2, 3, and 4, and which in turn invoke ANSI Standard N45.2 and its ancillaries directly as shown in Figure 3.1. Some of these ANSI Standards are presently invoked in an oblique fashion through the regulatory guides identified in SRP Chapter 17. However, notable exceptions are ANSI Standards N45.2.14 (now IEEE Standard 467), N45.2.15, and N45.2.16.

Further problems arise because other applicable ANSI Standards are not invoked in the SRP and could be very useful if applied to the RPS and RHR. Notable among these are ANSI/ASME NQA-1 which is a logical ancillary to 10 CFR 50, Appendix B, ANSI N18.8/ANS 4.1, and ANSI/ANS N51.1 which would establish requirements for reliability-related sub-elements in the design assurance TLR.

Two 10 CFR 50, Appendix A, general design criteria which are relevant to the RPS design requirements in cases involving anticipated operational occurrences are GDC 10, "Reactor Design," and GDC 15, "Reactor Coolant System." Neither of these requirements is identified in SRP Chapters 7, 13, 14, and 17 but both are identified in Chapter 15. This will assure that a basis for evaluating the functional reliability of the RPS during an anticipated operational occurrence is included in the appropriate SRP chapters.

SRP chapters applicable to the RPS and RHR do not invoke 10 CFR 50, Section 50.72, "Notification of Significant Events," and its ancillaries, Regulatory Guide 1.16, NUREG-0161 and 10 CFR 21. Since the Licensee Event Report (LER), together with 10 CFR 21, "Reporting of Defects and Noncompliance," is used by the NRC Office for Analysis and Evaluation of

Operational Data (AEOD) and by Inspection and Enforcement (IE) as their primary sources of operating data, it is important that these criteria be invoked as the basis for acceptance of the experience feedback elements of a reliability assurance program during the operations phase.

It is noted that a rule change regarding the LER has been proposed as 10 CFR 50, Section 50.73, which (among other things) would endorse the Institute for Nuclear Power Operations (INPO) plan to assume responsibility for the management, funding, and technical direction of the Nuclear Plant Reliability Data System (NPRDS). This rule change, when implemented, should also become a basis for evaluating the acceptability of the experience feedback element of a reliability assurance program.

Although AEOD and IE are not presently inferring reliability from data collected under the aegis of 10 CFR 50.72, other organizations identified in Reference 5 are using a larger data base (including the LER) to obtain data for reliability, availability, and maintainability (RAM) analyses.

IEEE Standards that are reliability-related are also obliquely invoked in the SRP. IEEE Standard 352, "Reliability Analysis of Nuclear Power Generating Stations," is invoked via IEEE Standard 379, Regulatory Guide 1.53, and Chapter 7 of the SRP. IEEE Standards 323 and 344, which are Class 1E environmental and seismic qualification standards, are invoked by Chapter 3 of the SRP and Regulatory Guides 1.89 and 1.100, respectively. Neither of these regulatory guides is invoked in SRP Chapters 7, 13, 14, and 17; however, they are invoked in Chapter 3. Since qualification data are relevant to system availability and experience feedback elements of reliability they should also be referenced in Chapter 17 in order to assure that the source of data enters the reliability data base.

ISA Standards ISA-S67.04 and ISA-dS67.06, which are relevant to operational reliability and experience feedback, are ancillary references in Regulatory Guide 1.105, Revision 2 (Task IC 010-5), and a draft regulatory guide (Task IC 121-5), respectively. The former has been reviewed and is in final form. Approval is expected in mid-1983. Comments on the latter have been received and are being reviewed. When these regulatory guides receive approval they will be invoked in the SRP.

An RDT Standard that is presently uninvoked in the SRP, but is highly relevant to reliability assurance, is RDT Standard F9-2T. This standard, although developed by the AEC for use in the liquid metal fast breeder reactor program, is applicable to light-water reactors and should be considered as the basis for an industry standard or regulatory guide or both.

Table 3.1 QUALITY ASSURANCE AND RELIABILITY ASSURANCE STANDARDS AND CRITERIA FOR REACTOR PROTECTION SYSTEMS AND RESIDUAL HEAT REMOVAL SYSTEMS

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY PPS	RHR
1. 10 CFR 50	Domestic Licensing of Production and Utilization Facilities							
a. 50.34	Contents of Applications; Technical Information	13	X			1	X	X
50.34(a)(7)	QA Program Description	17	X			1	X	X
50.34(b)(6)(i)	Organization and Personnel Qualifications	14		X	RG 1.8	1, 2, 3, 4	X	X
50.34(b)(6)(iii)	Plans for Preoperational Testing	14		X		1, 3	X	X
50.34(b)(6)(iv)	Plans for Operations, Maintenance, Surveillance, and Periodic Testing	13 18		X X		4	X	X
50.34(b)(8)	Plans for Operator Requalification	13 18		X X		4	X	X
b. 50.55a(h)	Criteria for Protection Systems for Nuclear Power Generating Stations (IEEE Std. 279)	7 17	 X	X	IEEE 279, WASH 1284 (Note 2), GDC-24	2	X	X
c. 50.55(e)	Conditions of Construction Permits (Reporting Significant QA Deficiencies)	17	X	X		2, 3	X	X
d. Appendix B	Quality Assurance Criteria For Nuclear Power Plants and Fuel Processing Plants	14 17		X X	RG 1.30, WASH 1309 (Note 3)	1, 2, 3, 4	X	X

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
e. 50.40(b)	Standards For Licenses and Construction Permits, Common Standards	13		X		1, 4	X	X
f. Appendix A	General Design Criteria For Nuclear Power Plants	17	X			1, 2, 3, 4		
GDC 1	Quality Standards and Records	7 17	 X	X		1, 2, 3, 4	X	X
GDC 10	Reactor Design	15		X		1, 2	X	
GDC 13	Instrumentation and Control	7		X		4	X	X
GDC 15	Reactor Coolant System	15		X		1, 2	X	X
GDC 18	Inspection and Testing of Electric Power Systems	8		X		4	X	X
GDC 19	Control Room	5		X		4		X
GDC 21	Protection System Reliability and Testability	7		X		1, 3, 4	X	X
GDC 23*	Protection System Failure Modes	7		X		1, 2	X	X
GDC 24	Separation of Protection and Control Systems	7		X	10 CFR 50.55a(h)	1, 2	X	X
GDC 29**	Protection Against Anticipated Operational Occurrences	7 15		X X		1, 2	X	

*Requires FMEA
**Requires conformance to PRA criteria

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
GDC 32	Inspection of Reactor Coolant Pressure Boundary	5		X	ANSI/ASME, B&PV Code, Section X1, 50.55a(g)	4		X
GDC 34	Residual Heat Removal	5		X		4		X
GDC 35	Emergency Core Cooling	6		X		4		X
GDC 36	Inspection of Emergency Core Cooling System	6		X		4		X
GDC 37	Testing of Emergency Core Cooling System	6		X		4		X
g. 50.55a(g)	Codes and Standards, Inservice Inspection Requirements (ASME B&PV Code, Section XI, 1971 Rules For Inservice Inspection of Nuclear Reactor Coolant Systems)	17		X	WASH 1283 (Note 4),	4	X	X
h. 50.72	Notification of Significant Events				NUREG-0161	4	X	X
i. 50.73	Licensee Event Report System (proposed rule change)				Not Presently Invoked by NRC	4	X*	X*
j. 50.59	Changes, Tests and Experiments					4		X
k. 50.36	Technical Specifications					4	X	X

*If Approved

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
1. 50.90	Application for Amendment of License or Construction Permit					4	X	X
2. 10 CFR 55	Operators Licenses; and its Appendix A, Requalification Programs For Licensed Operators of Production and Utilization Facilities	13	X		WASH 1284	4	X	X
3. ANSI Standards								
a. N45.2 - 1977	Quality Assurance Program Requirements for Nuclear Power Plants	17	X		WASH 1309, WASH 1283, WASH 1284 RG 1.28, RG 1.30, RG 1.33, RG 1.70	1, 2, 3 4	X	X
b. N45.2.1	Cleaning of Fluid Systems and Associated Components During the Construction Phase of Nuclear Power Plants	17	X		WASH 1309, RG 1.37	3		X
c. N45.2.2	Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase)	17	X		WASH 1309, RG 1.38	3		
d. N45.2.3	Housekeeping During the the Construction Phase of Nuclear Power Plants	17	X		WASH 1309, RG 1.39	3		

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
e. N45.2.4	Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	17	X		RG 1.30, IEEE-336 WASH 1309	3	X	X
f. N45.2.5	Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	17	X		WASH 1309, RG 1.94	3	X*	X*
g. N45.2.6	Qualifications of Inspection, Examination and Testing Personnel for the Construction Phase of Nuclear Power Plants	17	X		WASH 1309, RG 1.58	3	X	X
h. N45.2.8	Supplementary QA Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants	17	X		WASH 1284, WASH 1309 (N45.2.8, Draft 3) RG 1.116	3	X	X
i. N45.2.9	Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants	17	X		RG 1.88	3. 4	X	X

*Applicable to Safety Class Structures for Class 1E Equipment

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
j. N45.2.10	Quality Assurance Terms and Definitions	17	X		WASH 1309, RG 1.74	1	X	X
k. N45.2.11	Quality Assurance Requirements for the Design of Nuclear Power Plants	17	X		WASH 1309, RG 1.64	1, 2	X	X
l. N45.2.12	Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants	17	X		WASH 1309, RG 1.144	1, 2, 3, 4	X	X
m. N45.2.13	Quality Assurance Requirements for Control of Procurement of Equipment, Materials and Services for Nuclear Power Plants	17	X		WASH 1309, RG 1.123	2	X	
n. N45.2.14	Quality Assurance Requirements for the Design and Manufacture of Class 1E Instrumentation and Electric Equipment for Nuclear Power Generating Stations				WASH 1309, IEEE 467	2	X	X
o. N45.2.15	Requirements for the Control of Hoisting, Rigging, and Transporting of Items at Nuclear Power Plant Sites				WASH 1309	3	X	X
p. N45.2.16	Supplementary Quality Assurance Requirements for the Calibration and Control of Measuring and Test Equipment Used in the Construction and Maintenance of Nuclear Facilities				WASH 1309	3, 4	X	X

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
q. N45.2.17	Quality Assurance Requirements for Control of the Welding Process for Nuclear Power Plants				WASH 1309	3		X
r. ANSI/ANS-3.1 (ANSI N18.1-1977)	Selection and Training of Nuclear Power Plant Personnel	13 17		X X	WASH 1284, RG 1.8	1, 4	X	X
s. ANSI N18.7/ ANS-3.2	Standard for Administrative Controls for Nuclear Power Plants	13 17		X X	WASH 1284, RG 1.33	1, 4	X	X
t. N101.4	Quality Assurance for Protective Coatings Applied to Nuclear Facilities	17	X		RG 1.54, WASH 1309	3		
u. ANSI/ASME NQA-1-1979	Quality Assurance Program Requirements for Nuclear Power Plants with Addenda -1a-1981 and -1b-1981				Not Presently Invoked by NRC	1		
v. ANSI N18.8/ ANS 4.1	Criteria for Preparation of Design Bases for Systems That Perform Protective Functions in Nuclear Power Generating Stations				Not Presently Invoked in NUREG-0800; Invoked in IEEE 308	1		
w. ANSI/ASME BPV-XI	ASME Boiler and Pressure Vessel Code Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components-Division 1	5		X	10 CFR 50, Section 50.55a(g)	4		X

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
4. Branch Technical Positions								
a. BTP CMEB 9.5-1	Guidelines for Fire Protection For Nuclear Power Plants	17	X		10 CFR 50, GDC 3	1, 2	X*	X*
b. BTP RSB 5-1	RHR Pump Testing	5	X			4		X
c. BTP ICSB 22	Periodic Testing	7	X		GDC 21, RG 1.22	4	X	X
5. Regulatory Guides								
a. RG 1.8	Personnel Selection and Training (Endorses ANSI/ANS-3.1, Section 4.6.1)	17	X	X	WASH 1284	1, 4	X	X
b. RG 1.22	Periodic Testing of Protection System Actuation Functions	7	X			4	X	X
c. RG 1.26	Quality Group Classifications and Standards for Water-, Steam- and Radioactive-Waste-Containing Components of Nuclear Power Plants	17	X	X		1, 2	X	X
d. RG 1.28	Quality Assurance Program Requirements (Design and Construction) (Endorses ANSI N45.2)	17		X	WASH 1309	1, 2, 3	X	X

*Applicable to CMF/CCF Analysis.

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
e. RG 1.29	Seismic Design Classification	17	X	X		1	X	X
f. RG 1.30	Quality Assurance Requirements for Installation, Inspection and Testing of Instrumentation and Electric Equipment (Endorses ANSI N45.2.4)	14	X		WASH 1309,	3, 4	X	X
g. RG 1.33	Quality Assurance Program Requirements (Operation) (Endorses ANSI N18.7/ ANS 3.2)	13 17	 X	X 	WASH 1284	1, 4	X	X
h. RG 1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants (Endorses N45.2.1)	14 17	X X		WASH 1284, WASH 1309, WASH 1383	3		
i. RG 1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants (Endorses N45.2.2)	17	X	X	WASH 1309, WASH 1383, WASH 1384	3		
j. RG 1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Endorses N45.2.3)	17	X		WASH 1309, WASH 1284	3		
k. RG 1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants (Endorses N101.4-1972)	17	X		WASH 1309	3		

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
l. RG 1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (Endorses N45.2.6)	17	X	X	WASH 1309, WASH 1284	2, 3, 4	X	X
m. RG 1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants (Endorses N45.2.11)	17	X	X	WASH 1283, WASH 1284	1, 2	X	X
n. RG 1.68	Preoperational and Initial Startup Test Program for Water-Cooled Power Reactors	13 14 17	X X X	 X	WASH 1284	3	X	X
o. RG 1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants	7	X			3		
p. RG 1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	7	X			3	X	X
q. RG 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants	13	X		NUREG 0800 Introduction	1, 2	X*	X*
r. RG 1.74	Quality Assurance Terms and Definitions (Endorses N45.2.10)	17	X		WASH 1309, WASH 1283	1	X	X

*Applies to FMEA

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
s. RG 1.88	Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records (Endorses N45.2.9)	17	X		WASH 1309 (N45.2.9, Draft 15)	2, 3, 4	X	X
t. RG 1.94	Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Endorses N45.2.5)	17	X		WASH 1309 (N45.2.5, Draft 3)	3	X*	X*
u. RG 1.105 Proposed Rev. 2 (Task IC 010-5)	Instrument Setpoints (Endorses ISA-S67.04)	7	X			2, 4	X	X
v. RG 1.116	Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems (Endorses N45.2.8)	14 17	X X		WASH 1309, WASH 1284	3, 4	X	X
w. RG 1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (Endorses N45.2.13)	17	X	X		2	X	X
x. RG 1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants (Endorses N45.2.12)	17	X	X	WASH 1309	1, 2, 3, 4	X	X

*Applicable to Safety Class Structures for Class 1E Equipment

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
y. RG 1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants (Endorses N45.2.23)	17	X	X		1, 2, 3	X	X
z. RG 1.16	Reporting of Operating Information - Appendix A Technical Specifications				NUREG-0161	4	X	X
aa. RG 1.118	Periodic Testing of Electric Power and Protection Systems	7	X		IEEE 338	4	X	X
bb. RG 1.53	Application of Single Failure Criterion to Nuclear Power Plant Protection Systems	7	X		IEEE 379	1, 2, 3, 4	X	X
cc. RG 1.89	Qualification of Class 1E Equipment for Nuclear Power Plants	3		X	IEEE 323	1, 2	X	X
dd. RG 1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants	3		X	IEEE 344	1, 2	X	X
ee. RG 1.149	Nuclear Power Plant Simulators for use in Operator Training	13		X	10 CFR 55	4		X
6. RDT Standards								
a. F2-9T	Reliability Assurance				Not Presently Invoked by NRC	1, 2		
b. F2-2	Quality Assurance Program Requirements				Not Presently Invoked by NRC	1		

DOCUMENT	TITLE	NUREG-0800 (SRP)			INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY	
		CHAPTER	G	A			RPS	RHR
c. F2-4T	Quality Verification Program Requirements				Not Presently Invoked by NRC	1		
7. IEEE Standards								
a. IEEE-352	Reliability Analysis of Nuclear Power Generating Stations				IEEE 379	1, 2		
b. IEEE-323	IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations	3		X	RG 1.89	1, 2	X	X
c. IEEE-344	Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations	3		X	RG 1.100	1, 2	X	X
d. IEEE-338	IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection System	7	X		RG 1.118	3, 4	X	X
e. IEEE-379	IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems	3	X		RG 1.53	1, 2	X	X
f. IEEE-467	Quality Assurance Requirements for the Design and Manufacture of Class 1E Instrumentation and Electric Equipment for Nuclear Power Generating Stations (Replaces ANSI N45.2.14)	17		X	ANSI N45.2.14	2		

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
g. IEEE-336	Standard Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations	17	X		ANSI N45.2.4	3	X	X
8. MIL Standards								
a. MIL-HDBK-217D	Reliability Prediction of Electronic Equipment				Not Presently Invoked by NRC	1, 2		
9. ISA Standards								
a. ISA-S67.04	Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants				RG 1.105, Proposed Rev. 2 (Task IC 010-5)	3, 4	X	X
b. ISA-dS67.06	Response Time Testing of Nuclear-Safety-Related Instrument Channels in Nuclear Power Plants				Draft REG Guide (Task IC 121-5)	3, 4		
10. 10 CFR 21	Reporting of Defects and Non-compliance				Not Presently Invoked in NUREG-0800	2, 3, 4	X	X
11. NUREG 0737	Clarification of TMI Action Plan Requirements							
a. I.A.1.1	Shift Technical Advisor	13		X		4		X
b. I.A.2.1	Immediate Upgrade of RO and SRO Training and Qualifications	13		X		4		X

DOCUMENT	TITLE	NUREG-0800 (SRP) CHAPTER	G	A	INTERFACING DOCUMENTS	LIFE CYCLE (NOTE 1)	APPLICABILITY RPS	RHR
c. I.A.2.3	Administration of Training Programs	13		X		4		X
d. I.A.3.1	Revise Scope and Criteria for Licensing Exams	13		X		4		X
e. I.B.1.2	Evaluation of Organization and Management	13		X		4		X
f. II.3.4	Training for Mitigating Core Damage	13		X		4		X
g. II.E.4.2	Containment Isolation Dependability	7		X		4		X
h. II.K.3.21	Restart of Core Spray and Low Pressure Coolant Injection System	7		X		4		X
12. NUREG 0660	NRC Action Plan Developed as a Result of the TMI-2 Accident							
a. II.E.3.2	RHR Reliability	6		X		4		X
b. II.E.3.3	RHR Study of Shutdown Heat Removal Requirements	6		X		4		X

Notes:

1. Legend For Life Cycle Identification:
   (1) Conceptual Design
   (2) Design Development/Procurement
   (3) System Integration (including construction, installation, and preop testing)
   (4) Operations/Maintenance/Periodic Testing

2. WASH 1284, Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants, 21 Oct 1973.

3. WASH 1309, Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants, 10 May 1974.

4. WASH 1283, Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants, Revision 1, 24 May 1974.

Table 3.2 CURRENT NRC AUDIT MATRIX FOR NUCLEAR SAFETY AND RELIABILITY ASSURANCE PROGRAM RELEVANT TO REACTOR PROTECTION SYSTEMS AND RESIDUAL HEAT REMOVAL SYSTEMS

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
1. Conceptual Design (contd.)	·Management (contd.)	·Organization Planning	Preliminary Safety Analysis Report (PSAR): ·Organization Descriptions ·interface Identification ·Organization Location and Independence ·Responsibility and Authority Definition ·Lines of Communication	1a, 1d, 3a, 3s	·Construction Permit Safety Evaluation Report (CPSER)	NRR: ·Quality Assurance Branch (QAB) ·Procedures and Test Review Branch (PTRB) ·Human Factors Engineering Branch (HFEB) ·Power Systems Branch (PSB) ·Instrumentation and Control Systems Branch (ICSB) ·Equipment Qualification Branch (EQB) ·Mechanical Engineering Branch (MEB) ·Containment Systems Branch (CSB) (contd.)	Conformance to SRP: ·Utility Accepts QA/RA Responsibility ·Work Delegation and Responsibility Defined ·Clear Management Controls and Communications ·Independence of Designer and QA/RA Organization ·Interface Control

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
1. Conceptual Design (contd.)	·Management	·Program Planning	PSAR: ·Program Description ·Implementation Methods ·Personnel Qualification	1b, 3a, 3k, 5m, 5d 5y	·CPSER	·Accident Evaluation Branch (AEB) ·Auxiliary Systems Branch (ASB) Others: ·Vendor, AE, Utility, QA/RA	Conformance to SRP: ·Commitment to QA/RA Preop Tests ·QA/RA Control of Computer Code Programs ·Identification of QA/RA Control of Fire Protection System
		·Program Review Planning	PSAR: ·Conceptual Design Review Methods and Procedures	1a, 1b, 1f	·CPSER	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Provisions to Verify Design Adequacy
		·Audit Personnel Qualification Planning	PSAR: ·Qualification Test Plans	5y, 3l, 5x	·CPSER	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Provision for Audits to Verify QA/RA Compliance

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
1. Conceptual Design (contd.)	·Design Assurance (contd.)	·Design Control Planning	PSAR: ·Scope Statements ·Provisions for Design Change ·Interface Controls ·Activities Plans ·Error Correction Procedure ·Design Review Procedures	3a, 3k, 5m, 1b, 1f	CPSER: ·Design Review Procedures ·List of Organizational Responsibilities ·Corrective Action Requirements ·Design Change Procedures ·Interface Procedures ·Design Review Procedures ·Procedures to Control Use of Computer Codes	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Disciplined Engineering Approach
		·Document Control Planning (contd.)	PSAR: ·Document Review and Approval Procedures ·Document Control Procedures .Records Control (contd.)	3m, 5w, 3i, 5s, 1f, 3a (contd.)	CPSER: ·Document Control Procedures ·Control Document Identification .Document Change Procedures (contd.)	·NRR: QAB ·Vendor, AE Utility, QA/RA (contd.)	·Provisions for Review and Approval of Documents (contd.)

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
1. Conceptual Design (contd.)	·Design Assurance (contd.)	·Document Control Planning	·Organizational Responsibilities Descriptions	3m, 5w, 3i, 5s, 1f, 3u	·Record Storage and Distribution Procedures	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Provisions for Review and Approval of Documents
			·Change Procedures				
		·Design Criteria	·Class 1E Seismic Design Requirements	3n, 3dd, 7c, 5e	·CPSER	·NRR: QAB, ICSB, MEB, PSB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
			·Class 1E Equipment Qualification Requirements	3n, 3cc, 7b, 5u	·CPSER		
			·Quality Group Classification	5c	·CPSER		
			·Reliability and Testability Criteria	1f (GDC-21)	·CPSER		
		·Failure Mode and Effects Analysis (FMEA)	·FMEA, Topical Reports	5q, 1f (GDC-23 and GDC-29), 5bb	·CPSER	·NRR: QAB, ICSB, MEB, PSB, AEB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
		·Human Factors Analysis	·PSAR (Chap. 18)	1a, 1b, 1f	·CPSER	·NRR: HFEB	·Conformance to SRP
		·Conceptual Design Review (Title I)	·Systems Design Description Drawings, Topical Reports	1a, 3v	·CPSER	·Vendor, AE, Utility, QA/RA	

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
1. Conceptual Design	·Design Assurance	·Preliminary Reliability and Maintainability (R&M) Specification	PSAR RPS R&M Requirements	1b, 1f (GDC-21) 3k, 5m	·CPSER	·NRR: QAB, ICSB, MEB, PSB ·Vendor, AE Utility, QA/RA	·Conformance to SRP
	·Component Availability	·System Quality/Reliability Level	·System Classification	5c, 5e, 3n	·CPSER	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
		·System Testing Methods and Compliance Planning	·Test Procedures	7b, 7c, 1a, 5cc, 5dd	·CPSER	·NRR: QAB, ICSB, MEB, PTRB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
	·Experience Feedback	·Corrective Action	·Plans for Reporting Discrepancies, Errors, and Changes	1c, 1d, 3a, 10	·CPSER	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
		·Data Recording and Distribution	·Plans for Records Storage and Distribution	1f (GDC-1)	·CPSER		·Conformance to SRP
2. Design Development and Procurement (contd.)	·Management (contd.)	·Organization	·PSAR Amendments	3l, 5x, 3a	·CPSER Supplements (If Required)	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
		·Program Implementation	·QA/RA Program	1d, 3a, 3m, 5d	·Audit Report		

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
2. Design Development and Procurement (contd.)	·Management	·Qualification Test Plans	·Qualification Test Plans	1a	·CPSER Supplement (If Required)	NRR: LQB	·Conformance to SRP
		·Program Review	·Implemented Review Procedure	1a, 3m, 5w	·CPSER Supplement (If Required)	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
		·QA/RA Auditor Training	·Audit Methods and Procedures	31, 5x, 5y	·Supplemental CPSER	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
		·Supplier Audit and Surveillance	·Designer's or Vendor's Shop	31, 5x	·Audit Reports	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
			·Equipment Qualification	7b, 7c, 5u	·Qualification Status Reports	·NRR: EQB, QAB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
	·Design Assurance (contd.)	·Revise R&M Specification	·Revised RPS R&M Requirements	1b, 1f, 3k 5m	·CPSER Supplement (If Required)	·NRR: QAB, ASB, ICSB, MEB, PSB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
		·Human Factors Analysis	·Revised Analysis	1a, 1b, 1f	·CPSER Supplement (If Required)	·NRR: HFEB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
2. Design Development and Procurement (contd.)	·Design Assurance	·FMEA	·Revised Analysis	5g, 1f (GDC-29)	·CPSER Supplement (If Required)	·NRR: QAB	·Conformance to SRP
		·Installation Testing and Inspection	·Procedures	3e, 5f	·CPSER Supplement (If Required)	·NRR: QAB, ASB ICSB, RTRB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
		·Single Failure Analysis	·Analysis, Topical Reports	1b, 5bb, 7e	·CPSER Supplement (If Required)	·NRR: QAB, ICSB, MEB, PSB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
		·Design Review (Title II)	·Revised Systems Design Description	1a	·CPSER Supplement (If Required)	·NRR: QAB, ICSB, MEB, PSB ·Vendor, AE, Utility, QA/RA	·Independent Verification and Validation of Design
		·Standards and Guidance Conformance	·FSAR	1f, 3a, 5c	·CPSER Supplement (If Required)	·NRR: QAB, ICSB, MEB, PSB Vendor, AE, Utility, QA/RA	·Conformance to SRP
	·System Availability	·Quality/ Reliability Levels	·Measured QA/RA Levels	5c, 5e	·CPSER Supplement (If Required)	·NRR: QAB, ICSB, MEB, PSB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
2. Design Development and Procurement	·System Availability	·System Testing Methods and Compliance	·Qualification Test	3cc, 3dd, 7b, 7c	·CPSER Supplement (If Required)	·NRR: QAB, EQB, ICSB, MEB, PSB ·Vendor, AE, Utility, QA/RA	·Conformance to SRP
	·Experience Feedback	·Corrective Action	·Defect and Noncompliance Reports, LERs ·QA Deficiency Reports ·Engineering Change Notices ·QA/RA Records	1c, 1d	·Audit Report	·IE ·NRR: QAB, ICSB, MEB, PSB ·Vendor, AE, QA/RA	·Conformance to SRP
		·Data Recording and Feedback	·Qualification Reports	5cc, 5dd, 7b, 7c	·Evaluation of Qualification Reports	·NRR: EQB, QAB ·Vendor/AE: QA/RA, Equipment Testing	·Conformance to SRP
3. System Integration, Including Construction, Installation and Preop Testing (contd.)	·Management (contd.)	·Organization Revision (contd.)	·Final Safety Analysis Reports (FSAR) ·Organization Charts and Descriptions ·Interface Identification (contd.)	1a, 1d, 3a (contd.)	·Evaluation of Organizational Revision in SER (contd.)	·NRR: QAB ·Vendor/AE: QA/RA (contd.)	·Conformance to SRP (contd.)

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
3. System Integration (Including Construction, Installation and Preop Testing) (contd.)	·Management	·Organization Revision	·Activities Index ·Organizational Responsibility	1a, 1d, 3a	·Evaluation of Organizational Revision in SER	·NRR: QAB ·Vendor, AE QA/RA	·Conformance to SRP
		·Program Planning Revision	·FSAR ·Program Descriptions	1a, 1d, 3a, 5d	·Evaluation of Program Revisions in SER	·NRR: QAB ·AE, Utility, QA/RA	·Conformance to SRP
		·Program Review	·Implemented Review Procedures	1a, 1d, 3a	·Evaluation of Program Review Methods in SER	·NRR: QAB ·AE, Utility, QA/RA	·Conformance to SRP
		·Supplier Audit and Surveillance	·Construction Site Activities ·Vendors Field Installations	3l, 5x 3c, 3f, 3o, 5k, 5d, 5h, 5j	·Audit Reports ·Inspection Reports	·IE ·NRR: ICSB, MEB, PSB ·Utility QA/RA	·Conformance to SRP
		·Qualification of Audit, Inspection, Examination and Testing Personnel	·Qualification Tests	3g, 5l, 5y	·SER	·NRR: QAB, OLB, LQB ·Utility, QA/RA	·Conformance to SRP
		·Management Personnel Indoctrination and Training	·R Indoctrination Program Plan	3s, 5g	·SER	·NRR: LQB, OLB, QAB ·Utility, QA/RA	·Conformance to SRP

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
3. System Integration (Including Construction, Installation and Preop Testing) (contd.)	·Management	·Preoperational Test Planning	PSAR: ·Preoperational Test Plans	1a, 5n	·CPSER	·NRR: QAB ·Vendor, AE, Utility, QA/RA	·Description of Preop Tests
	·Design Assurance	·Installation, Inspection and Test of Mechanical Systems and Safety Class Structures	·Inspection and Test Reports	3h, 3f, 5t, 5v	·SER	·IE ·NRR: MEB RSB, QAB, ASB ·Utility, QA/RA	·Conformance to SRP
		·Installation, Inspection and Test of Instrumentation and Electrical System	·Inspection and Test Reports	3e, 5f	·SER	·IE ·NRR: ICSB, PSB, QAB ·Utility, QA/RA	·Conformance to SRP
		·Preop Testing	·Test Results	5n, 5o, 5p	·SER	·NRR: QAB, ICSB, MEB, PSB ·Utility, QA/RA	·Conformance to SRP
		·Human Factors Analysis	·Control Room Design	1a, 1b	·SER	·NRR: HFEB, ICSB ·AE, Utility, QA/RA	·Conformance to SRP
		·FMEA Revision	·Revised FMEA	5q, 1f (GDC-23 and GDC-29)	·SER	·NRR: ICSB, MEB, PSB	·Conformance to SRP
		·Single Failure Analysis Revision	·Revised Analysis	5bb	·SER	·NRR: ICSB, MEB, PSB	·Conformance to SRP

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
3. System Integration (Including Construction, Installation and Preop Testing)	·Design Assurance	·Conformance to Standards and Guidance	·FSAR	1e	·SER	·NRR: QAB	·Conformance to SRP
	·Operating Reliability Assurance	·Operations Personnel Indoctrination and Training	·Written Examinations	2, 3r, 5a	·SER	·NRR: LQB, OLB ·Utility, QA/RA	·Conformance to SRP
			·Accident Sequence Diagnosis and Prognosis Training Procedures	2	·SER		
			·FSAR	1a	·SER		
	·Component Availability	·Measurement and Test Equipment Control	·Procedures	3p	·SER	·NRR: QAB ·Utility, AE QA/RA	·Conformance to SRP
	·Experience Feedback	·Corrective Action	·Defect and Nonconformance Reports ·Preop Test Reports ·QA/RA Records ·Engineering Change Notices	1c, 1d, 3a, 10 5z, 5f	·SER	·AEOD ·IE ·NRR: QAB, ICSB, MEB, PSB	·Conformance to NUREG-0161 and SRP Requirements
		·Data Recording	·Preop Test Reports on RPS/ESFAS	5n	·SER	·NRR: QAB, ICSB, MEB ·Vendor, AE, QA/RA	·Conformance to SRP

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
4. Operations, Including Maintenance and Periodic Testing (contd.)	·Management	·Periodic Program Review and Revision	·Revised QA/RA Program	1a, 5x, 11e	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Requalification Plan Review	·Revised Requalification Plan	1a	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·QA Personnel Indoctrination and Training	·Revised Plans	1d	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Test Procedure Review and Revision	·Revised Test Procedures	1j	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Training Program Review and Revision	·Revised Training Program	2, 5y, 11c	·IE Evaluation Report	·IE QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Operations Program Requirements Review and Revision	·Revised Operations Program	3a, 5ee, 3s, 5q, 11a	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Assessment of Program Effectiveness	·IE Audit Report	3l, 5x	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Mangement Personnel Training and Revision	·Revised Training Plans	3r, 5a	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
4. Operations, Including Maintenance and Periodic Testing (contd.)	·Design Assurance	·Design Modification Description and Change Notices	·Modification Description	1f (GDC-13, 19, 34, 35), 1j	·IE/NRR Safety Evaluation Report	·IE: QAB ·Regional Office ·Utility QA/RA ·NRR	·Conformance to Licensing Conditions and Commitments
		·Major Design Modification Description	·Application for Amended License	1l	·IE/NRR Safety Evaluation Report	·IE: QAB ·Regional Office ·Utility QA/RA ·NRR	·Conformance to Licensing Conditions and Commitments
		·Design Modification Analysis	·Report	5bb, 7e	·IE/NRR Safety Evaluation Report	IE: QAB ·Regional Office ·Utility QA/RA ·NRR	·Conformance to Licensing Conditions and Commitments
		·R Study	·Report	12a	·IE/NRR Safety Evaluation Report	·IE: QAB ·Regional Office ·Utility QA/RA ·NRR	·Conformance to Licensing Conditions and Commitments

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
4. Operations, Including Maintenance and Periodic Testing (contd.)	·Design Assurance	·Performance Study	·Report	12b	·IE/NRR Safety Evaluation Report	·IE: QAB ·Regional Office ·Utility QA/RA ·NRR	·Conformance to Licensing Conditions and Commitments
	·Operations R (contd.)	·Periodic Program Review and Revision	·Program Revisions	1a	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Periodic Requalification	·Qualification Reports	1a	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Periodic Testing	·Test Reports	1f (GDC 18, 21, 36, 37), 1k, 3w, 4b, 4c, 5b, 5aa, 7d	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Periodic Inspection	·Inspection Reports	3w, 4b	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Personnel Qualification (contd.)	·Qualification Lists (contd.)	2, 5a, 5l, 5y, 5ee	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
4. Operations, Including Maintenance and Periodic Testing (contd.)	·Operations R (contd.)	·Operations and Maintenance Personnel Training	·Training Programs	3r, 5a	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Operator Requalification	·Requalification Reports	11b, 11d, 11f, 11g 11h	·IE Evalution Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Start-up Testing	·Test Reports	5p, 5n, 5o	·IE Evaluation Report	·IE:QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
	·Experience Feedback (contd.)	·QA Records and Audit Reports	·Report	1d, 5a	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Design Change Notices	·Notices	1f (GDC 1)	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Test Reports	·Reports	1f (GDC 18, 32, 37), 5n, 5o, 5p, 5aa, 7d, 9b	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Inspection Reports	·Reports	1f (GDC 36)	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments

LC Category	TLR Program Elements	Program Subelements	Auditables	Requirements Documentation	Compliance Documentation	Organization Interfaces	Metrics
4. Operations, Including Maintenance and Periodic Testing (contd.)	·Experience Feedback (contd.)	·Event Reports	·Notification of Events	1h, 1i	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Evaluation Reports	·Design, Test or Experimental Change Descriptions	1j	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Test and Calibration Records	·Technical Specification Surveillance Documents	1k	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Quality Records	·QA Documents	3a	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Calibration Reports	·Calibration Records	5u, 9a	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Licensee Event Reports	·Report	5z, 1i, 1h	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments
		·Defect and Noncompliance Reports	·Report	10	·IE Evaluation Report	·IE: QAB ·Utility QA/RA	·Conformance to Licensing Conditions and Commitments

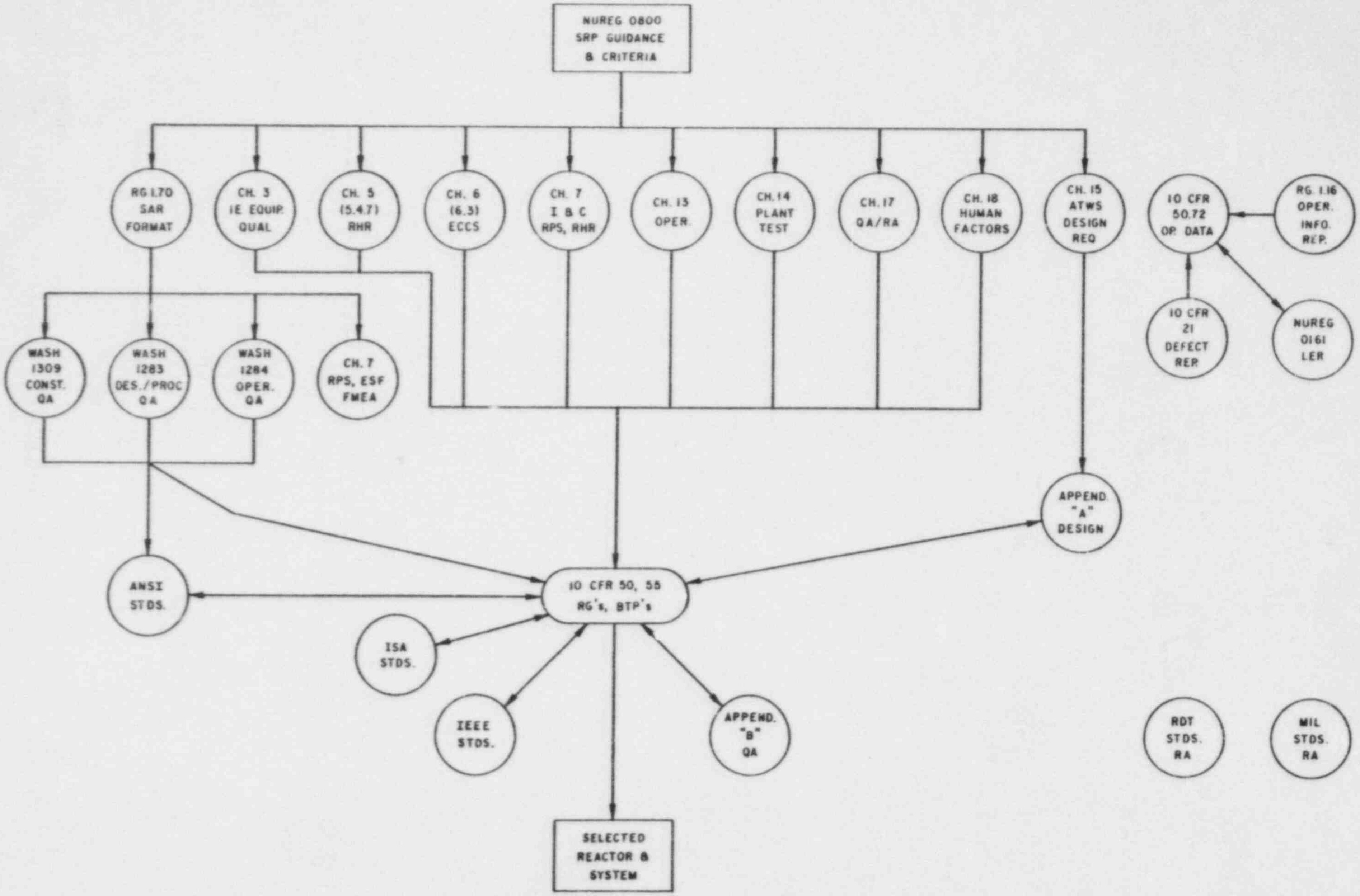

Figure 3.1 NRC QA/RA Document Interfaces-BF1 RPS & RHR

4. Regulatory Practices Relevant to RHR Reliability

The RHR in its normal operational mode and in its emergency (LPCI) mode is required to meet the guidance and criteria identified in Table 3.1 in the previous chapter. This chapter discusses, within the context of the top level reliability program requirements (TLRs) of Table 3.2, current NRC practices in regulating the safety and reliability of the RHR during the "operations" life cycle phase.

During the operational phase of a nuclear power reactor the primary responsibility for the determination of the conformance of the licensee to NRC quality and reliability-related criteria and guidance is assigned to the NRC Office of Inspection and Enforcement (IE)[a]. IE uses NUREG-0800 criteria and guidance, which are used by the Office of Nuclear Reactor Regulation (NRR) as the bases for acceptance of the reactor during the licensing review, as the gauge of the effectiveness of the quality and reliability assurance programs. The documents invoked in NUREG-0800 which are related to safety and reliability assurance of the RHR and LPCI are listed in Table 3.1. If the organization and program effectiveness are found to be deficient then the IE Quality Assurance Branch (QAB) and the licensee's quality assurance organization cooperate in reviewing and revising the program to meet required goals.

The scope of current NRC requirements, which are applicable to the operations phase, presently includes four primary quality and reliability assurance program elements. These program elements are management, design assurance, operations reliability assurance, and experience feedback. Of the documents listed in Table 3.1, 18 are related to management, 11 to design assurance, 30 to operations reliability, and 24 to experience feedback. These operations-phase TLRs and reliability program sub-elements are further identified in Table 3.2.

4.1 Management Aspects

Table 3.1 lists the criteria and guidance which are related to the management reliability program elements that provide the bases for reviewing,

[a]As noted in Appendix B of this report, certain organizational changes within NRC are under way. Many of the responsibilities of IE are being transferred to NRC regional offices.

evaluating, and revising the program to either correct administrative deficiencies discovered in the process of implementing the planned quality and reliability assurance programs or to meet changing needs caused by revised requirements. These changes can be implemented where necessary in the following program sub-elements.

Organization: The implementation of the QA organization plan, accepted by the NRC during the operating license review, is monitored by IE. This includes all persons and organizations involved in operations, maintenance, program review, design change control, procurement activities, document control, test control, corrective action, records, audits, and the training of QA and RA personnel.

Organizational requirements are defined in 10 CFR 50, Appendix B, and in Regulatory Guide 1.33 and its ancillary Standards ANSI N45.2 and N18.7. Appendix B requires that the quality assurance organization identify quality and reliability problems and initiate, recommend, or provide solutions to these problems.

These requirements are intended to assure that the management organization can respond to changing needs by controlling and revising the implemented programs as necessary during the operational lifetime of the plant.

Program and Program Review: Requirements for quality assurance and reliability assurance programs during the operations phase of a nuclear power plant (including maintenance, testing, and plant modification) are defined by 10 CFR 50, Section 50.34(b)(6), Section 50.34(b)(8), Appendix B, and 10 CFR 55 and by ancillary ANSI Standards N45.2 and N18.7, and Regulatory Guide 1.33. Appendix B, Regulatory Guide 1.8, and ancillary ANSI Standard N18.1 require that quality and reliability programs provide for indoctrination and training of personnel performing activities related to quality and reliability assurance and that the programs be audited or reviewed regularly to determine status and adequacy, and revised if necessary.

Program auditing requirements are defined in Regulatory Guide 1.144 and ancillary ANSI Standard N.45.2.12. Periodic evaluation of the organization and management is also required by NUREG-0737, Item I.B.1.2.

Personnel Training: In current practice the operational safety and reliability are assured by requirements for an adequately trained and qualified staff. Requirements for the selection, indoctrination, and training of personnel for nuclear power plants are described in Regulatory Guide 1.8 and ancillary ANSI Standard N18.1. The indoctrination and training of administrative and management personnel are included in those criteria and supplemented by ANSI Standard N18.7.

The selection, training and qualification of audit personnel are also within the purview of the administrative organization. Requirements are described in Regulatory Guide 1.146 and ancillary ANSI Standard N45.2.23.

The periodic review and evaluation of the effectiveness of the administrative and audit staff training programs required by Appendix B, are intended to assure that the programs remain relevant.

4.2 Design Assurance

The reliability of the RHR is presently assured by requirements for periodic testing, inspection, and calibration and by the control of design or procedural changes related to those systems.

Inspection, Test and Surveillance: Requirements for verification of the integrity, status, and availability of the RHR by inspection and test are defined in 10 CFR 50, Appendix A, GDCs 18, 21, 36, and 37, and Section 50.55a(g), and by periodic tests of the Class 1E electric power system, and protection systems such as the RHR performed in accordance with the requirements of 10 CFR 50, Section 50.55a(h) (IEEE Standard 279, Section 4.10), Regulatory Guide 1.118, and IEEE Standard 338.

Regulatory Guide 1.118 provides guidance for determining conformance to 10 CFR 50, Section 50.55a(h), GDC 21, and IEEE 338. It assures operations reliability by requiring periodic testing and inspection of the RHR (LPCI mode) to confirm operational availability. At the present time there is no guidance in Regulatory Guide 1.118 concerning the reporting of a failure detected during testing that could form the basis for experience feedback. However, its ancillary Standard IEEE 338 recommends that failure data be collected in a recognized failure data collection bank such as the Edison Electric Institute or the Nuclear Plant Reliability Data System (NPRDS). IEEE 338 also describes the bases for periodic tests including applicable reliability modeling, reliability allocation, failure modes and effects, and failure report analysis.

The results of inspections and tests provide the basis for inferred reliability of the system or can, in the event of failures, form the basis for revisions in system design criteria through the data provided in accordance with 10 CFR 50, Section 50.72, Regulatory Guide 1.16, and 10 CFR 21.

Design and Procedural Change Controls: Design assurance is also afforded by the present NRC policy which prohibits uncontrolled and unreviewed changes from being made in the RHR design or in procedures related to those systems.

If the licensee desires to make a change in the facility design or procedures previously accepted by the NRC or to conduct tests and experiments not previously defined, it is required by 10 CFR 50, Section 50.59, that the licensee demonstrate to the NRC that such changes or tests do not involve unreviewed safety questions. Such changes are documented by the licensee and reviewed by IE and NRC regional offices. The system which is changed must conform to the same General Design Criteria that were used as the bases for accepting the original design.

Any changes requiring a revision of technical specifications or involving previously unreviewed safety issues require that the licensee apply for an amendment of the license in accordance with 10 CFR 50, Section 50.90.

Additional guidance is provided by Regulatory Guide 1.105 and its ancillary Standard ISA-S67.04 in requirements for the determination of set point accuracies and limits. Each instrumentation channel is required to be calibrated and functionally tested at intervals specified in the technical specifications. This also includes the testing of bypass and interlock logic and response time testing.

Analysis: As a result of the lessons learned from TMI-2, the NRC Office of Nuclear Reactor Regulation (NRR) plans to conduct a generic study (NUREG-0660, Item II.E.3.2) to assess the capability and reliability of shutdown heat removal systems under various transients and degraded plant conditions including complete loss of all feedwater. Deterministic and probabilistic methods will be used to identify design weaknesses and possible system modifications that could be made to improve the capability and reliability of these systems under all shutdown conditions (i.e., start-up, hot standby, shutdown, etc.).

The NRR is also planning to conduct a coordinated effort to evaluate shutdown heat removal requirements thereby permitting a judgment of adequacy in terms of overall system requirements. As part of this effort, NRR will conduct a study to assess the desirability of and possible requirement for a diverse heat-removal path, such as feed and bleed, particularly if all secondary-side cooling is unavailable.

However, these efforts are incomplete at this time and the issue of shutdown decay heat removal requirements is unresolved (see Reference 6).

4.3 Operations Reliability Assurance

Operational reliability of the RHR is presently assured by start-up testing, operator training and qualification, and periodic testing, inspection, and surveillance requirements as follows.

Start-Up Testing: The operational reliability of the RHR is presently assured by the results of start-up testing required by Regulatory Guide 1.68 and its ancillary Regulatory Guide 1.68.2.

In addition to providing evidence that the system operates within the technical specification limits, these tests serve to give the operating and plant technical staff practical experience with the reactor and an opportunity to trial-test plant operating and emergency procedures for the RHR including remote shutdown.

Periodic Testing, Inspection and Surveillance: Operations reliability is assured by the 10 CFR 50.55a(g) periodic in-service inspection and test requirement for the mechanical portions of the RHR as augmented by the requirements of 10 CFR 50, Appendix A, GDC 36 and 37. Similarly, GDC 18, Regulatory Guide 1.22, Regulatory Guide 1.118, Regulatory Guide 1.105, and IEEE Standard 338 also provide criteria and guidance for the assurance of reliability of the electrical systems by periodic testing of the RHR.

Present test and surveillance practices for BF1 RHR systems include tests for pump operability, MOV operability, flow rate, and drywell and torus spray. Additionally, periodic tests of the RHR pumps are required by Branch Technical Position BTP 5-1.

During operation IE monitors the licensee's implementation of plans for operation, maintenance, and periodic testing in accordance with the requirements of 10 CFR 50, Section 50.34(b)(6)(iv), and Section 50.36, "Technical Specifications," and verifies conformance to the qualification requirements for inspection, examination and testing personnel as defined by Regulatory Guide 1.58.

At the present time requirements for the response time testing of the instrumentation channels associated with the automatic isolation of the RHR from the recirculation system are not invoked in NUREG-0800. However, a draft regulatory guide which may endorse the approved final version of the draft ISA Standard dS67.06 could alter this.

Operating Personnel Training and Qualification: Operations reliability is presently assured by requirements for personnel selection, training, qualification, and requalification of reactor operators in accordance with the plans required by 10 CFR 50, Section 50.34(b)(8), 10 CFR 55, and ancillaries Regulatory Guide 1.8 and ANSI Standard 18.1.

In addition, NUREG-0737, "Clarification of TMI Action Plan Requirements," requires immediate upgrading of reactor operators and senior reactor operator training and qualifications (Item I.A.2.1), a revision in the scope and criteria of licensing exams (Item I.A.3.1), training for mitigating core damage (Item II.B.4), clear operator identification of essential and nonessential systems (Item II.E.4.2), and restart of core spray and LPCI systems (Item II.K.3.21). IE also will check for conformance to these NUREG-0737 criteria.

4.4 Experience Feedback

Experience feedback during the operations phase is required by 10 CFR 50, Section 50.72, and its ancillary Regulatory Guide 1.16. The former requires that the licensee notify the NRC through the Licensee Event Report (LER) of the occurrence of significant events including the reporting of personnel errors or procedural inadequacies which could impair the functioning of the RHR.

It is also required by 10 CFR 21 that defects in the RHR discovered during operation be reported in writing which discloses the nature of the defect or failure and the safety hazard which is created or could be created by such a defect. In the case of the failure of basic components, the number and location of all such components in use at the facility or similar facilities must also be reported.

Information concerning the results of start-up tests including the testing of the RHR and the demonstration of remote shutdown in accordance with the guidance of Regulatory Guides 1.68 and 1.68.2 would be contained in the test reports and would also be reported to the NRC in accordance with 10 CFR 21 in the event defects or deficiencies are found.

The results of periodic tests which are required by 10 CFR 50, Section 50.55a(g) and Appendix A, GDC 18, 36 and 37, and ancillary Regulatory Guide 1.118, are recommended for collection in NPRDS or the EEI data collection bank by IEEE Standard 338.

4.5 Discussion

A review of Tables 3.1 and 3.2 indicates that the current NRC requirements are relatable to five principal TLR areas. These are reliability management, design assurance, component availability, operations reliability, and experience feedback. These TLRs are comparable to the tentatively recommended nuclear systems reliability program identified in Reference 7. However, additional reliability sub-elements which are relevant to the operational reliability of a nuclear reactor, notably those sub-elements related to operator selection, training, and qualification, and human reliability in general, appear only in the regulatory requirements. Comparable industry reliability program elements are not identified in Reference 7.[b]

The unique NRC emphasis on operational reliability is apparent in the analysis of Table 3.2 where the reliability program sub-elements and invoked documents related to operational reliability were dominant.

The reliability-related regulatory requirements and guidance identified in Table 3.1 are not presently incorporated by the NRC into a formal reliability assurance program. Therefore, while the potential to audit the reliability-related program elements listed in Table 3.2 presently exists, such reliability-oriented audits have yet to be identified specifically in the SRP.

It is not possible to say what the effects of implementing a reliability assurance program based on these criteria and guidance would be. However, one may speculate that the existing criteria for the design of stationary boiling water reactor plant systems, such as the BF1 RPS and RHR, would have explicit reliability requirements imposed on them during the conceptual and detailed design phases. Further, one can assume that these reliability requirements could be imposed in lieu of other quality group or safety class requirements such as those identified in R.G. 1.26 and ANSI N51.1. The demonstration of system reliability and its acceptance by the NRC may then modify other requirements, such as those associated with operations reliability including periodic inspection and testing during the operations life cycle phase of the plant as described herein.

Any justifiable relaxation of technical specification surveillance requirements based on reliability demonstration would certainly be welcomed by the industry because of its relationship to economics. No attempt is made

[b]It is the objective of the study described in Reference 7 to develop basic criteria, requirements and application guidelines for defining reliability program elements and tasks based on an independent study of the nuclear and non-nuclear power generating industry as well as other organizations such as DOD, NASA, and the FAA.

here to speculate on the overall impact of a reliability assurance program on the operational process. However, a discussion of possible impacts on the licensing process is given in Chapter 2.

5. Preliminary Conclusions and Future Efforts

As reported in the previous chapters a review and evaluation of the current regulatory requirements and guidelines in the area of reliability have been made for two typical safety systems in a boiling water reactor. The initial goal of this subtask was to benchmark the current safety- and reliability-assurance-related practices employed by the NRC, utility licensee, and system designer/vendor, focusing on specific systems important to safety. Two representative safety systems, the reactor protection system (RPS) and the residual heat removal system (RHR) of the Browns Ferry Unit 1 BWR, were used as the reference systems in this appraisal. Current regulatory requirements for these systems were condensed into a matrix identifying the corresponding auditable elements of a reliability assurance program (RAP) to establish a basis for comparative evaluation with RAPs from other high technologies. This evaluation will indicate the current regulatory requirements and industry practices that could be integrated with or replaced by elements of a formal reliability assurance program more relevant to safety and reliability, more conducive to the licensing process, and more easily auditable by the NRC in the operational phase of a plant.

Reliability related standards and criteria found in the "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants" (Ref. 1) and Title 10 of the Code of Federal Regulations (10 CFR) formed the primary basis of this study. The SRP is essentially a compendium of acceptable solutions to meeting the requirements in 10 CFR. By checking all references in the appropriate SRP chapters for the reference RPS and RHR systems, the regulatory guides, NRC branch technical positions, and industry consensus standards having implications for a RAP were identified and catalogued. The completed catalogue (Table 3.1) of RAP-relevant standards and criteria describes where the standards/criteria are invoked, the major interfaces with supporting documents, and their applicability over each life cycle phase of a nuclear plant.

Using as a model the reliability assurance programmatic elements recommended by IITRI and the Rome Air Development Center in a previous NRC study, the standards and criteria identified above were then organized into these elements. The resulting aforementioned matrix of current regulatory requirements organized by life cycle phase, top level reliability assurance function, and material directly auditable by the NRC, provides the requirements benchmark for comparison with the RAPs from the FAA, NASA, and DOD. It also provides the baseline for integrating or replacing current requirements with a RAP-oriented licensing approach. It is concluded that the implementation of a RAP would not necessitate new requirements so much as it would require the folding in of existing requirements into a more systematic and readily auditable format specified by a reliability assurance program.

Also undertaken as a part of this task was a brief review of the licensing process to indicate how and where NRC implementation of a performance-oriented RAP could be used to reduce licensing efforts and schedules. This licensing review with indications of the potential RAP impacts on licensing is reported in Chapter 2. FAA evolution toward regulating performance rather than the details of achieving performance (e.g., design details) has strongly enhanced regulator-industry cooperation in the aircraft certification (analogous to licensing) process and is judged to have enhanced safety. It is clear that use of a quantitative performance-oriented RAP in licensing in conjunction with safety goals could also reduce the plant-to-plant variability in safety posture implied by the spread in current PRA results by imposing risk-oriented availability requirements on safety-related systems. Finally, a detailed summary of the current (non-reliability-based) licensing process is also provided as Appendix A to allow subsequent in-depth comparisons with the FAA approach, the analysis of which is part of other phases of the RAP research program.

To benchmark the actual NRC practices in monitoring safety system operations, a summary review of current NRC practices relevant to reliability assurance of the residual heat removal system in the operations phase of the life cycle was included. Reduction of NRC monitoring requirements is another objective of RAP. To facilitate the consolidation of RAP requirements within the NRC, a summary description of the various NRC organizational units that currently have responsibilities in reliability-related efforts has been included as Appendix B.

Because of the difficulties in obtaining the necessary information for review of the licensee and vendor practices, the output described above is limited to the current NRC requirements and practices that relate explicitly or implicitly, to requirements of a RAP. The proper completion of this task requires the summarization of how the industry assures the reference system safety, reliability, and regulatory compliance. This step is considered to be a small but necessary effort in the integration of a RAP with current requirements and requires the input of an operating utility. However, based upon the results of the analysis of reliability assurance within the regulatory process conducted so far it is concluded that:

1. The NRC presently has, within the existing regulatory requirements, the potential for auditing many recognized reliability and maintainability program elements that exist in the nuclear industry; however, an audit of these elements has not been incorporated into current SRP licensing requirements.
2. Although reliability is explicitly called for in many of the standards and regulatory guidelines found in the SRP, it is usually used in a qualitative sense. Only in rare cases does reliability imply a quantitative reliability criteria. None of the invoked documents describes a reliability program to the degree of detail found in 10CFR50, Appendix B for a quality assurance program.
3. Although a quality assurance program is invoked in the present NRC licensing process, reliability assurance programs requiring analytical techniques (e.g., failure mode and effects analysis, common mode/common cause failure analysis, single failure analysis and fault tree analysis) are not required for obtaining licensing approval but are nevertheless routinely practiced by much of the industry.
4. The results of this work compiled in Table 3.2 showed that within the current body of NRC rules, requirements and guidance, a framework already exists in which to integrate, not add, a reliability-based regulatory program. This is a very important conclusion, for unless elimination of some current requirements can be guaranteed, there is no way imposition of RAP requirements will be accepted by the industry in a non-adversial way. They will perceive RAP as another mechanism for ratcheting additional requirements.
5. No general correlation of the present NRC reliability audit capabilities, discussed in this report, with the reliability assurance programs presently used by the utility (TVA) and (GE) the designer/vendor could be

made at this time because of the lack of adequate reliability-specific operational documents for the BF1 RPS and RHR.

Future reliability benchmarking efforts planned to extend the range of this analysis include a review and assessment of the utility and the designer/vendor reliability programs. This will allow a correlation to be made between the cause (i.e., regulatory requirements) and effects (i.e., licensee operation) and further identify reliability assurance program elements for the RPS and RHR safety systems. The framework for factoring in industry practices has been established in this preliminary phase and it should be a reasonably small effort to include this industrial component. However industry support and cooperation is required in this endeavor.

In later tasks of this program a systemmatic comparison will be made between the reliability assurance programs identified from review of selected FAA, NASA and DOD programs and the reference reliability program elements presently invoked in the nuclear industry. Based upon the results of these comparisons and the reliability needs as determined by the risk-related requirements obtained in another task of this research, a RAP will be developed. Essential to this process of integrating or replacing current requirements with a RAP approach will be the question of costs versus potential benefits. As indicated above unless elimination of some current requirements can be guaranteed, the industry will resist implementation of a RAP. The experience base, both in reliability performance and cost improvement, as obtained in the other high-technology industries will be relied on to provide the necessary data and information essential in the support of a comprehensive RAP nuclear regulatory approach.

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Appendix A

The NRC Regulatory Process

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The NRC Regulatory Process

This appendix describes the current NRC process used in the review of applications for reactor construction permits and operating licenses, as well as the post-operating license surveillance activities. The following material is mostly excerpted from "Licensing Project Manager's Handbook", U. S. Nuclear Regulatory Commission, December 30, 1977 (Rev. 8). A number of NRC organizational changes have been made since publication of this handbook. In addition, certain responsibilities of HQ groups described herein are being transferred to NRC regional offices. Because these transfers are in transition at this time, and because the basic process is not affected by these changes, no attempt has been made to correct the relevant material. Finally, there are several active proposals for overhauling and streamlining the licensing process, including both legislative and administrative initiatives. Since none of those proposals are expected to substitute RAP-based licensing for the present mode, we do not discuss them further here.

A.1 NRC Licensing Review

The licensing review of each application for a nuclear power plant is accomplished in two principal phases: the CP (construction permit) phase and the OL (operating license) phase. In the application for a CP, the applying utility delineates the approach which it intends to take for the design and construction of the proposed nuclear facility.

Review by the Office of Nuclear Reactor Regulation (NRR) is directed toward the determination of the potential for a safe and environmentally acceptable plant within the criteria and preliminary design information presented by the applicant.

During the review of the application for an OL, the steps in the process are similar, but the application is reviewed from the viewpoint of a more complete design and operational type information. The OL review concerns itself with the implementation of the criteria and preliminary design information to arrive at the final design in a manner consistent with protection of public health and safety and minimal environmental impact.

A.1.1 Construction Permit

The initial contact in the submittal of an application is usually by the utility at higher management levels, followed by a meeting with the staff to discuss proposed sites, type of plant and other characteristics, and the considerations that will be applied in the review process. The utility then prepares an application for submittal, usually in concert with the selected architect-engineer and the nuclear steam supply system supplier. In the event the site selected is judged to be controversial, the applicant may elect to initially submit information for a site review only. The tendered application for a CP consists of the application itself containing general information (10 CFR 50.33) including schedule and financial qualifications; the Preliminary Safety Analysis Report (PSAR) containing the supporting technical information (10 CFR 50.34); and/or the Environmental Report (ER, 10 CFR 51). The application can be submitted with either the ER or SAR being furnished up to six months later as a separate submittal. In these cases, separate acceptance reviews will be conducted. The information required for an antitrust review is submitted 9-36 months prior to the application and SAR or ER submittal.

Within a period of 30 days following receipt of a tendered application, the Office of Nuclear Reactor Regulation conducts an acceptance review of the application for completeness. The acceptance review is managed by the responsible licensing project manager (LPM), with the various chapters of the PSAR or ER reviewed by the cognizant review branches and consultants, and the general and financial information reviewed by the licensing assistant in consultation with the financial analyst. The appropriate LPM coordinates the review, and makes an overall assessment of the degree of completeness of the submitted portions of the application based on input from the various reviewers, and develops a coordinated recommendation to management with respect to the acceptability of the application for review.

When the application is tendered, it is made available to the public in the Public Document Rooms (PDRs) and as soon as the application is docketed, the application and supporting documents are distributed to the staff, as well as to the Advisory Committee on Reactor Safeguards (ACRS), and both the

Technical Information Center and the Nuclear Safety Information Center at the Oak Ridge National Laboratory. The documentation is also sent to the Council on Environmental Quality, appropriate federal agencies, state, local and other appropriate officials. If the application is judged acceptable for docketing, the applicant is notified and informed regarding the informal process available for him to discuss areas of dispute that may arise from staff positions. Shortly after docketing, Federal Register notices are issued, announcing the receipt of the application and that a public hearing will be held. The hearing notice provides an opportunity for interested persons to petition for leave to intervene, and also indicates that a special prehearing conference will be scheduled (normally within 60 days of acceptance of the application for docketing). The appropriate LPM, having developed an approved project schedule, and the cognizant review branches and consultants initiate the review of the application.

Shortly after the end of the intervention period (30 days following publication in the Federal Register) the LPM initiates discussion with potential intervenors and takes the initiative to arrange a meeting with them to discuss the nature of their contentions. If possible, this meeting is held just prior to the first prehearing conference.

The first prehearing conference usually establishes intervention status and sets the ground rules for discovery. Contentions of the intervening parties, at least for purposes of prehearing procedures, should be established at this first prehearing conference.

When the review has progressed to the point at which a number of concerns have been identified and documented in draft form, a meeting is arranged between the staff and their counterparts in the applicant's organization in order to discuss the areas requiring elaboration and to further identify and define the issues. These concerns are formulated as technical questions by each reviewer and, in turn, are reviewed, coordinated and assembled by the LPM for transmittal to the applicant as a request for additional information. In some cases, where time is of the essence, the LPM may also elect to utilize phone contact and follow with the documentation later. If a meeting is deemed desirable to discuss certain questions, they may be sent to the applicant in draft form as a meeting agenda.

During the review process, the applicant has the prerogative to request an exemption pursuant to 10 CFR 50.12 to permit certain site preparation and construction activities to proceed prior to granting of a CP. Current policy, however, is that these exemptions will be granted very sparingly and the scope of the work authorized will not exceed that authorized by a limited work authorization (LWA) until the environmental review is complete.

After the safety review is complete, and the necessary additional information has been documented by the applicant in the PSAR, the LPM prepares the Safety Evaluation Report (SER) from input provided by the various review branches. The SER is sent to the ACRS and is also made public at this time.

When the environmental review is complete, the Draft Environmental Statement (DES) is prepared by the environmental project manager (EPM), published and transmitted to federal, state, and local agencies for comment, and to parties to the hearing for information. Notice of availability of the DES is published in the Federal Register.

The Final Environmental Statement (FES), incorporating the responses to comments received from the review of the DES, is then published.

If the applicant desires an LWA pursuant to 10 CFR 50.10(e), the Atomic Safety and Licensing Board (ASLB) will schedule hearings on environmental and site suitability matters as soon as practicable after issuance of the FES and will issue a separate initial decision on environmental and site suitability matters. During this period, the EPM will initiate staff action for issuance of an LWA. The LPM manages the site suitability aspects of the LWA review and must concur in the issuance of the LWA. Accordingly, the LPM is expected to be aware of any significant problem areas that have developed during the technical evaluation of the application that could have a bearing on the issuance of an LWA.

The PSAR is reviewed by the ACRS. The ACRS meets with the applicant and the staff initially at the subcommittee level and then as a full committee to discuss the key safety issues identified during the review of the project as well as other issues the Committee believes are of importance. Members of the public may attend and make statements at these meetings. The ACRS advises the Commission, as required by law, with respect to the conclusions of its independent review. This advice, in the form of an ACRS letter addressed to the Chairman of the Commission, is available to the public through the PDR and a press release.

Following the ACRS review, a supplemental SER must be prepared and issued. The supplemental SER consists primarily of the ACRS letter and the staff's response to the comments contained therein, as well as any other pertinent information that needs to be documented in the public record that was not available at the time the SER was published.

At about this time another prehearing conference is usually held in order to establish the schedule and format for the conduct of the public hearing.

The public hearing is usually held near the proposed site. The principal participants at the hearing from the staff include the LPM, review personnel and consultants if necessary.

A public hearing on the issuance of a CP is required in all cases. Following the hearing, an Initial Decision is rendered by the ASLB. As discussed above, separate decisions may be issued on environmental and safety issues. If the decision is favorable, a CP is issued. The Initial Decision is subject to review by the Atomic Safety and Licensing Appeal Board (ASLAB) and the Commission.

Issuance of the CP by the Director of NRR, including receipt of a water quality certification from the cognizant agency, appropriate antitrust review, and payment of the CP fee are the usual necessary and sufficient conditions to permit construction of the facility to begin under the general terms of the permit.

After the CP has been issued, any items of review that remain outstanding and that cannot reasonably be left for the OL review, should be resolved on a determined schedule while final design and construction of the plant proceed.

Although the objective of the CP review is to eliminate the number of outstanding items, a very few items may still remain for consideration during this two or three year post-CP period. To resolve these outstanding items, the applicant submits additional documented information in the form of an amendment to his application. The staff must review this information and inform the applicant by letter of the results and conclusions of the review. The appropriate LPM coordinates or conducts these reviews and assures that all matters appropriate to this phase of the overall review are completed.

A.1.2 Operating License

The procedural aspects of the review process for an OL are very similar to those for a CP. The content of the application reviewed, however, is substantially different. The applicant amends his original application by providing a Final Safety Analysis Report (FSAR) and submitting new or amended general information in the application. The FSAR is usually a completely new document based on the actual design and including operational plans, but consistent in format with the latest requirements for PSARs. The applicant must submit an Environmental Report for the Operating License Stage which discusses those items significantly different than previously reviewed during the construction permit environmental review. No further antitrust review is required unless significant changes in the situation have occurred since the previous review at the construction permit stage.

The initial stage, as in the case of a CP review, is an acceptance review of the tendered application to determine its acceptability relative to completeness. Formal and extensive distribution of the application does not occur until it is judged to be acceptable for review. The docket number continues as before, so that no new docket number is assigned.

The OL review is primarily directed toward a determination of the acceptability of translation of the design criteria and preliminary design information, specified during the CP review, into the final design and construction of the nuclear facility, and toward the review and evaluation of plans related to operation. Additional Commission requirements that may have been developed since issuance of the CP must also be factored into the OL review.

One of the major tasks in performing the OL review is the development of suitable Technical Specifications (Tech Specs). The Tech Specs become the portion of the OL that governs the subsequent operation of the facility relative to the health and safety of the public and to the protection of the environment. They identify and define all the limits and requirements that the licensee must abide by without change unless specific approval is obtained from the NRC. Development of the Tech Specs is accomplished in much the same way as the review and evaluation of the SAR itself. Input information is provided by the licensee, review and evaluation is performed by the respective review personnel, and the overall management and integration responsibility is carried out by the LPM. The schedule objective is to complete the development of the Tech Specs by the time the supplemental SER is issued.

In the past, Tech Specs were developed individually for each plant. They were similar for plants of like design but there was always quite a bit of tailoring and some unavoidable inconsistencies. Sufficient experience exists now such that Standard Technical Specifications have been developed for each vendor design. New operating licenses are being issued with Standard Tech Specs.

An OL hearing is not mandatory, but the regulations do give the public the opportunity to request a hearing. This is accomplished by publishing in the Federal Register a Notice of Receipt of Application, Notice of Consideration of Issuance of a Facility OL, Notice of Availability of Environmental Report and Notice of Opportunity for Hearing. If potential intervenors present their intention to raise certain issues, the Commission or the board appointed by the Chairman of the ASLB panel decides on the validity of the issues raised by the potential intervenors and determines whether a hearing should be held. The issues upon which a hearing would be structured must be based on specific grounds cited by the potential intervenors. Uncontested issues are not reviewed by the ASLB. If a hearing is held, the same general procedures apply with regard to pre-hearing conferences and scheduling of the public hearing as during the CP stage.

The OL is a license to "possess, use and operate" the nuclear facility. As part of the necessary approvals prior to the issuance of a license, the Office of Inspection and Enforcement (IE) must certify that the plant has been constructed in accordance with regulatory requirements and the design commitments made in the application. If these design commitments have changed since the construction permit was issued, the changes will necessarily have to have been reviewed and approved during the operating license review.

The CP contains an administrative limitation regarding the earliest and latest completion dates for the nuclear facility. An applicant must apply for an extension of the CP expiration date in the event the construction of the nuclear facility is not completed by the latest date specified in the CP. The OL, however, is effective over the design lifetime of the plant, but not more than forty years from issuance of the CP.

No specific plan for the decommissioning of the plant is required at the time of licensing. This is consistent with the Commission's current regulations which contemplate detailed consideration of decommissioning near the end of a reactor's useful life. The licensee initiates such consideration at that time by preparing a proposed decommissioning plan. However, decommissioning of the facility may not commence without authorization from the NRC.

The LPM continues his responsibility for reviewing the various safety aspects of the facility until the plant has been licensed for a significant power level or until most of the hearing matters have been concluded. At this time, responsibility is transferred to the operating reactor project manager (ORPM). The majority of the ORPM activity relating to operating power reactors is related to Tech Spec changes, modifications including refuelings, and changes in the analyses presented in the FSAR. The changes and modifications result from equipment or operating deficiencies which have occurred at one or more facilities, changes in basic parameters from those used in the FSAR, upgrading requirements of the NRC, or operating needs of the licensee.

A.2 Post-Operating License Activities

During the post-OL phase, the responsibility for monitoring the licensee's activities and performance, primarily at the facility location, rests with the Office of Inspection and Enforcement (IE).

Following the issuance of an OL, the LPM may maintain responsibility for the project for a relatively short period of time, although the goal is to transfer responsibility to the ORPM as soon as possible. Issues which the LPM must handle during this period may involve equipment problems, operational problems, Tech Spec changes, and abnormal occurrences as reported by the licensee and/or IE inspections. Normally, all plants are transferred to the Division of Licensing (DOL) at the time of licensing for a significant power level.

Although plants are transferred to DOL at the time of significant power licensing, the LPM might be responsible for one or more license amendments prior to the transfer. The need for a Tech Spec change or other license amendment is most frequently determined by the licensee as a result of certain difficulties during plant operation or a desire to improve plant performance. Technical justification for such proposed changes is submitted by the licensee to the NRC staff by letter for consideration. Other changes may be initiated by the NRC staff as a result of information obtained from the operation of other facilities or reviews of other applications. For either situation, the LPM may undertake the sole responsibility for the review of the appropriate information to determine the need for and acceptability of a specific change, or request the assistance of other NRC groups. Any Tech Spec change or other license amendment which the LPM handles during this period must be coordinated with the assigned ORPM and with the Standard Tech Spec Group Leader in DOL, if appropriate.

The change process may entail a question-response cycle, meetings with the applicant, and preparation and submittal of appropriate documentation. The NRC staff's evaluation and conclusions are transmitted to the licensee by letter prepared by the LPM and signed by his assistant director. One of the conclusions that must be reached involves whether the requested change raises an unresolved safety issue. If it does, an SER must be prepared.

Abnormal occurrences which arise during this phase of plant operation are normally the responsibility of IE. Specific items might be turned over to NRR by request of IE if the problem is generic or for some other valid reason.

Review and evaluation of the periodic operating reports submitted during the post-OL phase by the licensee in accordance with the Tech Spec requirements is also the responsibility of IE.

In summary, through its inspection and enforcement program the NRC maintains surveillance over construction and operation of a plant throughout its lifetime to assure compliance with Commission regulations for the protection of public health and safety and the environment.

Appendix B

Reliability Functions Within the NRC

Appendix B

Reliability Functions Within the NRC

In order to understand the current NRC philosophy regarding reliability assurance, it is informative to review both its structural organization and the functiors of its offices, divisions, and branches. The structure of the NRC is show herein. The function of its component parts is described in NUREG-0325 "U. S. Nuclear Regulatory Commission Functional Organization Charts," Revision 5, January 1, 1983.

This Appendix identifies those divisions and branches whose functional responsibilities involve them in either a principal or supporting role in current NRC reliability-related efforts. Figure A.1 shows the overall organization structure of the NRC as configured in early 1983 with a list of abbreviations for the reliability-related NRC offices, divisions and branches provided in Table A.1. Presented below are excerpts of the mandates for the NRC organizations which presently have a direct or supportive role in reliability. There may be other NRC organizations whose mandates may involve them indirectly in reliability-related efforts, however, these groups are not discussed herein. For further details on the function of these as well as all of the NRC group mandates NUREG-0325 should be consulted.

B.1 Office of Nuclear Reactor Regulation (NRR)

NRR develops and administers regulations, policies, and procedures governing the licensing of nuclear power plants and the licensing of the operators of such facilities.

Division of Safety Technology (DST)

The principal NRR division involved in reliability-related efforts is the Division of Safety Technology (DST). Its Reliability and Risk Assessment Branch (RRAB) assesses reliability of systems important to safety, develops reliability-related analytical techniques, and trains NRR reviewers in reliability methods. It is also involved in coordinating NRR/RRAB and the Office of Nuclear Reactor Research (RES) Reactor Risk Branch (RRB) activities as well as the Interim Reliability Evaluation Program (IREP).

At the present time DST and the Office of Nuclear Regulatory Research (RES) Division of Risk Analysis (DRA) are directly involved in reliability-related efforts and are mandated to coordinate in establishing reliability standards and guidance for branches to use in license reviews. DST's Research and Standards Coordination Branch (RSCB) provides the necessary coordination in those areas of mutual interest to NRR and RES.

Division of Systems Integration (DSI)

Several branches of DSI, notably the Instrumentation and Control Systems Branch (ICSB), Containment Systems Branch (CSB), Reactor Systems Branch (RSB), Power Systems Branch (PSB), and Auxiliary Systems Branch (ASB), are mandated to evaluate the conformance of a nuclear plant to the criteria and guidance invoked in NUREG-0800. Some of these criteria and guidance are related to known reliability program elements, e.g., those identified in Table 3.2 of this report.

However, little or no de facto evaluation of the conformance of the plants reviewed by NRR to specific reliability criteria and guidance established by RES is known to be required by NUREG-0800 at this time. If such requirements exist, they have not been documented.

Division of Engineering (DE)

Three branches of DE, the Mechanical Engineering Branch (MEB), the Equipment Qualification Branch (EQB), and the Quality Assurance Branch (QAB), are also involved in the evaluation of conformance to NUREG-0800 criteria and guidance.

The MEB establishes the seismic and quality group classification of mechanical systems and components and also makes independent risk evaluations. The QAB evaluates specific conformance to 10 CFR 50, Appendix B, criteria. The EQB evaluates the capability of plant systems and components important to safety to function under all anticipated conditions. EQB also establishes performance requirements and reviews qualification test programs and results.

Division of Human Factors Safety (HFS)

This division has four branches whose mandates are not presently clearly linked to reliability-related efforts of other branches. Nevertheless, the generic mandates deal with subjects that are reliability-related and are implicitly related to the subject of Human Reliability Analysis (HRA) which is an emerging technique in the field of nuclear power plant operational reliability. The branches involved are the Licensee Qualifications Branch (LQB), Human Factors Engineering Branch (HFEB), Operator Licensing Branch (OLB), and Procedures and Test Review Branch (PTRB).

HFS is the newest of the NRR divisions. As such, its role in reliability assurance is not yet established.

B.2 Office of Nuclear Regulatory Research (RES)

RES plans, recommends, and implements programs of nuclear regulatory research which the NRC deems necessary for the performance of its licensing and related regulatory functions.

Division of Risk Analysis (DRA)

The principal RES division involved in reliability-related efforts is DRA. Within DRA, the Reactor Risk Branch (RRB) has a mandate to prepare standards and regulations related to risk and reliability. It also assists in collecting reliability data.

A second branch, the Regulatory Analysis Branch (RAB), performs a supportive role. It reviews the results of NRC's and other research programs to identify regulation changes needed to meet new requirements, e.g., for a reliability assurance program, and to eliminate unnecessary regulatory constraints. This branch is not directly involved in reliability-related efforts of DRA.

Division of Facility Operations (DFO)

The DFO plans, develops, and directs research and standards programs for nuclear safety with an emphasis on human factors for all life cycle phases of a nuclear reactor. Its Human Factors Branch (HFB) provides a supportive role in the areas of human factors and quality assurance. This includes the safety-related aspects of the man-machine interface. The HFB activities require coordination with NRR/HFS and NRR/QAB as well as the Office of Inspection and Enforcement (IE). The HFB is not presently involved in the reliability-related efforts of DRA.

B.3 Office for Analysis and Evaluation of Operational Data (AEOD)

AEOD is responsible for the analysis and evaluation of operational safety data associated with all NRC-licensed facilities and is directly involved in the feedback of information during the operation phase of nuclear power plants and in the archiving of data. The principal data reviewed by AEOD are those which are required by 10 CFR 50.72 and 10 CFR 21. While reliability-related, these data are not presently in a form in which quantitative reliability information can be easily obtained.

The Reactor Operations Analysis Branch (ROAB) of AEOD performs the review, analysis, and evaluation of reactor operating experience from the viewpoint of improving public health and safety. The Program Technology Branch (PTB) develops and operates data systems for the storage and retrieval of operational experience and does trend and pattern analysis of that experience.

B.4 Office of Inspection and Enforcement (IE)

IE develops policies and programs to determine whether the licensees, applicants, and their contractors and suppliers are in conformance with applicable NRC rules, regulations, orders, and licensing conditions. Their mandate also includes an evaluation of quality assurance programs which are reliability-related.

The Program Support Branch develops and administers IE policies and procedures for personnel and training which assure that inspection and audit personnel are adequately trained.

Division of Reactor Programs

The IE inspection and training program is managed by the Division of Reactor Programs. Under this division are the Reactor Construction Programs Branch and the Operating Reactor Programs Branch. Each branch has a mandate to develop inspection programs in its respective area. These programs are relevant to design assurance elements of reliability efforts.

Division of Engineering and Quality Assurance

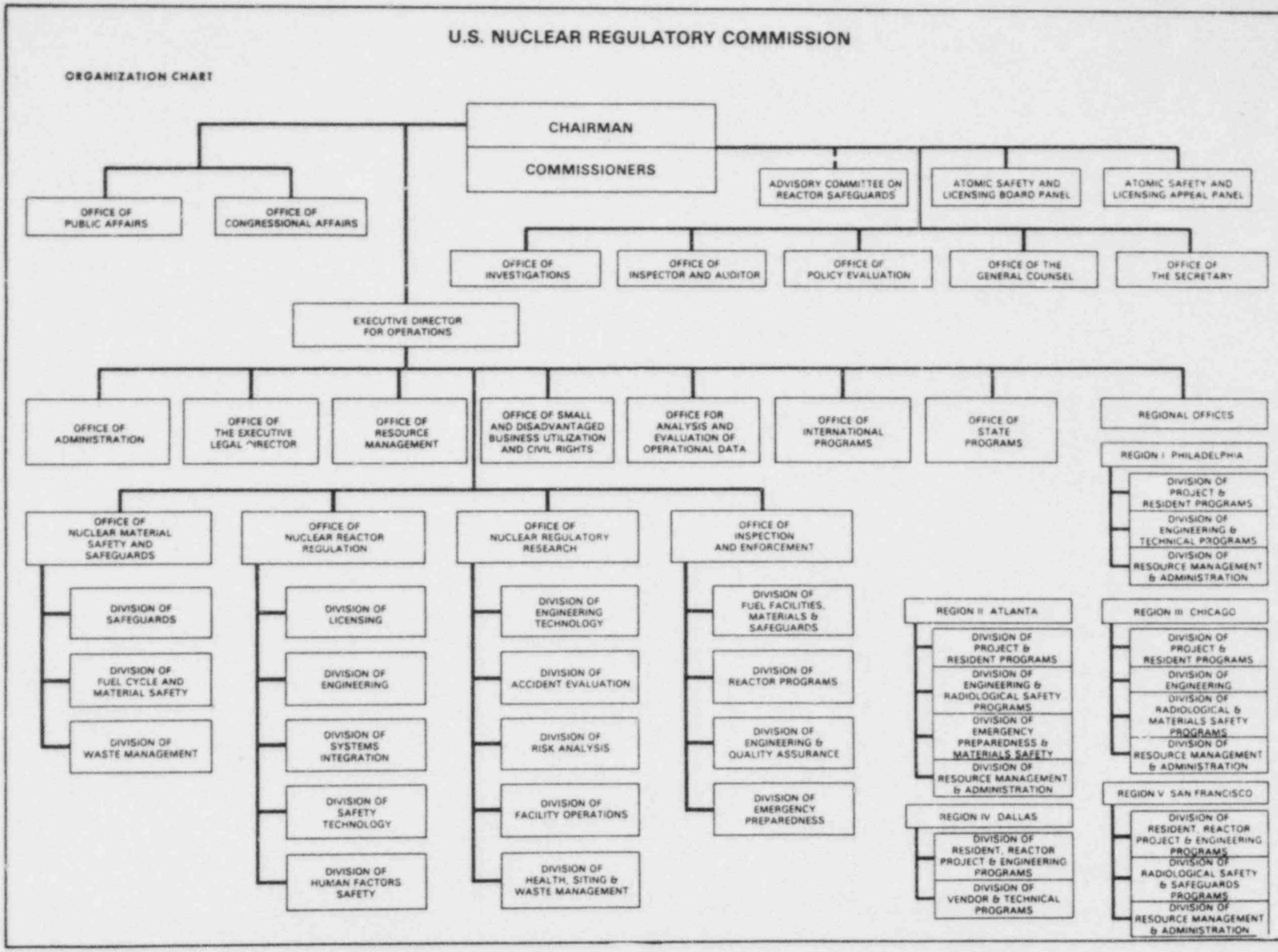
This division develops NRC policy and program requirements for the review of reactor events. Under this division the Events Analysis Branch (EAB) develops IE policy and prepares procedures regarding responses to events and communicates the results of analysis to other NRC offices. This effort is relevant to the information feedback element of a reliability assurance program.

The Quality Assurance Branch which is under this division (not to be confused with NRR/QAB) has a mandate to develop the NRC quality assurance policy and program for all phases of the reactor life cycle and would be responsible for making policy and program changes necessary to develop future reliability assurance programs. However, present quality assurance policy does not explicitly stress reliability.

Table B.1
Abbreviations* For Reliability-Related
NRC Offices, Divisions, and Branches

1. Office of Nuclear Reactor Regulation	NRR
1.1 Division of Safety Technology	DST
· Reliability and Risk Assessment Branch	RRAB
· Research and Standards Coordination Branch	RSCB
1.2 Division of Systems Integration	DSI
· Instrumentation and Control Systems Branch	ICSB
· Containment Systems Branch	CSB
· Reactor Systems Branch	RSB
· Power Systems Branch	PSB
· Auxiliary Systems Branch	ASB
1.3 Division of Engineering	DE
· Mechanical Engineering Branch	MEB
· Equipment Qualification Branch	EQB
· Quality Assurance Branch	QAB
1.4 Division of Human Factors Safety	HFS
· Licensee Qualifications Branch	LQB
· Human Factors Engineering Branch	HFEB
· Operator Licensing Branch	OLB
· Procedures and Test Review Branch	PTRB
2. Office of Nuclear Regulatory Research	RES
2.1 Division of Risk Analysis	DRA
· Reactor Risk Branch	RRB
· Regulatory Analysis Branch	RAB
2.2 Division of Facility Operations	DFO
· Human Factors Branch	HFB
3. Office for Analysis and Evaluation of Operational Data	AEOD
· Reactor Operations Analysis Branch	ROAB
· Program Technology Branch	PTB
4. Office of Inspection and Enforcement	IE
· Program Support Branch	IE/PSB
4.1 Division of Reactor Programs	PRB
4.2 Division of Engineering and Quality Assurance	DEQA
· Events Analysis Branch	EAB
· Quality Assurance Branch	IE/QAB

*Abbreviations used here may not be officially recognized by the NRC.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OFFICE OF PUBLIC AFFAIRS
OFFICE OF CONGRESSIONAL AFFAIRS
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
ATOMIC SAFETY AND LICENSING BOARD PANEL
ATOMIC SAFETY AND LICENSING APPEAL PANEL
OFFICE OF INVESTIGATIONS
OFFICE OF INSPECTOR AND AUDITOR
OFFICE OF POLICY EVALUATION
OFFICE OF THE GENERAL COUNSEL
OFFICE OF THE SECRETARY
EXECUTIVE DIRECTOR FOR OPERATIONS
OFFICE OF ADMINISTRATION
OFFICE OF THE EXECUTIVE LEGAL DIRECTOR
OFFICE OF RESOURCE MANAGEMENT
OFFICE OF SMALL AND DISADVANTAGED BUSINESS UTILIZATION AND CIVIL RIGHTS
OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA
OFFICE OF INTERNATIONAL PROGRAMS
OFFICE OF STATE PROGRAMS
REGIONAL OFFICES
REGION I PHILADELPHIA
DIVISION OF PROJECT & RESIDENT PROGRAMS
DIVISION OF ENGINEERING & TECHNICAL PROGRAMS
DIVISION OF RESOURCE MANAGEMENT & ADMINISTRATION
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
DIVISION OF SAFEGUARDS
DIVISION OF FUEL CYCLE AND MATERIAL SAFETY
DIVISION OF WASTE MANAGEMENT
OFFICE OF NUCLEAR REACTOR REGULATION
DIVISION OF LICENSING
DIVISION OF ENGINEERING
DIVISION OF SYSTEMS INTEGRATION
DIVISION OF SAFETY TECHNOLOGY
DIVISION OF HUMAN FACTORS SAFETY
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Figure B.1

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J. B. Heineman
J. M. Kramer
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Manager, Chicago Operations Office, DOE
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L. C. Hebel, Xerox Corp., 3333 Coyote Hill Rd., Palo Alto, Calif. 94304
S. Levine, NUS Corp., 910 Clopper, Gaithersburg, Md. 20878
W. F. Miller, Jr., Los Alamos National Lab., Los Alamos, NM 87545
M. J. Ohanian, U. Florida, Gainesville, Fla. 32611
T. H. Pigford, U. California, Berkeley, Calif. 94720
J. J. Taylor, Electric Power Research Inst., Palo Alto, Calif. 94303

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9700 South Cass Avenue
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13 ABSTRACT (200 words or less)
The objectives of this work are to evaluate and benchmark the current safety and reliability assurance-related practices employed by the NRC. This effort represents an initial phase of a program whose overall purpose is to develop a reliability program (RP). A review of NRC regulations relevant to reliability assurance was made for a boiling water reactor using two representative safety systems; the reactor protection system, and the residual heat removal system. The primary sources of information were the NRC standard Review Plan and Title 10 of the Code of Federal Regulations, especially Part 50. In addition, relevant regulatory guides, NRC branch technical positions and industry consensus standard were identified and catalogued for the two reference safety systems over the plant's life cycle. The identified standards and criteria were then organized into a RP element matrix of current regulatory requirements organized by life cycle phase, top level assurance function, and items directly auditable by the NRC. A brief review of the licensing process was also undertaken to indicate the effectiveness of NRC implementation of a RP. The results of this work showed that within the NRC regulations a framework already exists in which to integrate, not add, a reliability assurance program.

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