REPORT

Final (Draft)

for

EFFECTIVENESS OF

LWR REGULATIONS

IN LIMITING RISK

TASK 24

to

Regulation Development Branch

Office of Nuclear Regulatory Research

U.S. Nuclear Regulatory Commission

May 26, 1989

Battelle

FINAL REPORT (DRAFT)

on

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to

REGULATION DEVELOPMENT BRANCH
OFFICE OF NUCLEAR REGULATORY RESEARCH
U.S. NUCLEAR REGULATORY COMMISSION

MAY 26, 1989

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EXECUTIVE SUMMARY

This report identifies existing regulatory requirements which should be reexamined for possible elimination or modification without compromising safety. The reported work used a systematic technical approach to update earlier surveys. Knowledgeable NRC staff provided the principal input through structured interviews. During the interviews, NRC staff also identified many other potential modifications intended to update, clarify or expand existing regulatory authority. These candidates are also reported here. Considered alongside other ongoing regulatory activities, this list of candidates suggests a potential agenda for rulemaking which improves regulatory efficiency.

Two other general observations are noteworthy. First, there is a frequently expressed concern about several general aspects of the regulatory requirements, namely their currency, use of consistent terminology, and the initial operating conditions assumed for safety analysis. Second, it is apparent that formal rulemaking significantly lags the development and implementation of new regulatory positions in many cases. These factors introduce complexity and uncertainty in terms of exactly what requirements the licensees of future reactors are expected to satisfy.

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1. BACKGROUND

Efficient and effective regulation of the nuclear power industry is a mission of the Nuclear Regulatory Commission. Since 1984, the Commission's annual policy and planning guidance has explicitly reinforced this objective.

> "Existing regulatory requirements should be reviewed to see if some could be eliminated without compromising safety, safeguards or environmental protection."[1]

Guidance from the Executive Director for Operations directed that

> "RES should continue...based on a comprehensive evaluation of their risk significance to select existing regulations for reexamination. Regulations should be eliminated or modified that have marginal importance to safety or that have become obsolete by implementation of other approved staff practices."[2]

Battelle Memorial Institute is providing technical support to the Regulation Development Branch in carrying out this directive. This report presents recent results and discusses them within the context of prior related work.

RES responded initially to the guidance from the Commission and EDO through a combination of public outreach, internal surveys, and contract support. Beginning with a research plan and Federal Register notice in October 1984,[3] RES solicited suggestions regarding which specific regulatory requirements or associated regulatory positions should be reevaluated. A survey conducted by the Atomic Industrial Forum provided industry's input. NRC staff participated in a 1985 survey by RES[4]. Taken together, all of these viewpoints yielded the sizeable list of candidates presented in NUREG/CR-4330, Volume 1.[5a]

From this list RES selected seven areas for more thorough technical assessments:

- containment leak rate testing,
- BWR main steam isolation valve leakage control systems,
- fuel design safety review,
- postaccident sampling system,
- turbine missiles,
- combustible gas control, and
- charcoal filters.

The assessments involved value-impact analyses employing available data and insights from quantitative risk assessments. The results were reported in NUREG/CR-4330 Volumes 2 and 3.[6b,c]

A Program Advisory Group, composed of members from each major office in NRC, reviewed these assessments and concluded that certain requirements in the first two areas could be considered as having marginal importance to safety and that certain modification of these requirements could produce significant savings in resources without adversely affecting public health and safety. In the third area, certain requirements could also be considered as being of marginal importance to safety, but there appeared to be no significant cost savings in modifying these requirements. The remaining four areas were not reviewed. At that point RES proposed several specific relaxations to 10CFR50 Appendix J, Containment Leakage Testing. However, consensus within the NRC staff was not achieved and the proposed rulemaking did not materialize.

Meanwhile other initiatives directed toward more efficient regulation were evolving within the NRC. These initiatives do not necessarily require changes in the regulations. For example, plant technical specifications were frequently mentioned as a topic ripe for improvement. Working with industry, an NRC task force has progressed to the point of standardizing technical specifications, eliminating nonessential material, and reducing test and surveillance requirements during plant operation. The staff plans to develop a p... cy statement on technical specifications which reflects the results of the task force.

Another example involved analyses related to pipe breaks. Based on an improved technical understanding of physical phenomena associated with leakage before break, NRR relaxed its requirements through a generic letter.[6] This relaxation allowed licensees to eliminate analyses of pipe whip effects in locations having no particular driving force for an increased probability of rupture. This also allowed licensees to remove many pipe whip restraints. Remaining intact, however, were requirements to analyze effects of postulated pipe breaks in locations of relatively higher stress, higher fatigue or terminations.

The three years which have passed since earlier surveys were completed have allowed new information to be generated from research, regulatory and operating experience. Therefore, RES is updating the prior surveys of NRC staff, placing particular emphasis on 10CFR Part 50. Furthermore, the technical defensibility of this activity is being enhanced by employing a more systematic and comprehensive approach to screening the requirements. The screening method relies extensively on the engineering judgement of the NRC staff as structured through technical support provided by Battelle. This report describes the results of that screening.

Our current plan is to follow the screening process with a quantitative, risk-based evaluation of those candidate requirements which lend themselves to such an evaluation. Other candidates which may not be so amenable to a risk-based assessment, such as reporting or procedural requirements, will be evaluated more subjectively. The results of these evaluations, taken together with other considerations such as burdens on industry, burdens on NRC resources, and licensing reform, are intended to lead to specific recommendations concerning elimination or modification of nonessential regulatory requirements and related improvements in the regulations.

2. TECHNICAL APPROACH

The structured interview was the principal tool used to gather and organize information. Interviewees were selected so as to assure reasonably comprehensive and insightful coverage of all areas of reactor regulation. They were to draw upon their expertise in their particular area, their experience in regulation, their knowledge of regulatory requirements, and any other information at their disposal. Appendix A contains the agenda and the accompanying material for a typical interview. Included in the preparatory materials was a printout of regulatory positions extracted from the Standard Review Plan, Code of Federal Regulations, and Regulatory Guides relevant to the interviewee's interests.

Opening remarks clarified the purpose of the interview. The next step was to review suggestions from earlier work to determine whether and to what extent previously recommended changes had been acted upon. Interviewers then solicited comments on additional candidates for elimination or modification in the sense of the Commission's guidance. If candidates were identified, further discussion focused on the rationale behind the potential action and, in a few cases, the nature and extent of regulatory burden involved. Interviewees were then encouraged to discuss any other changes they believed were appropriate and the relevance of current regulatory positions to the regulation of future reactors. Notes from the interviews were reviewed and organized into the results provided in Section 3. The Glossary in Table 1 was developed to aid clarity.

To enhance comprehensiveness and systemization of the data gathering, Battelle developed a relational data base entitled REGIS, a REGulatory Information System. REGIS utilizes dBASE-IV™, commercially available data management software, operating on an IBM-compatible personal computer. dBASE-IV™ allows the user to input information into tabular format, and then search the data base. For this application, REGIS, Battelle used the Standard Review Plan[7] as a systematic and comprehensive compilation of regulatory positions that could serve as an anchor point for further efforts.

The data base contains the following information for each section of the Standard Review Plan:

- Section number and title
- Primary and secondary review branch
- References to 10CFR
- References to Regulatory Guides
- References to NUREG reports.

REGIS was used to help prepare for structured interviews and in several cases to serve as a means of identifying within minutes all the regulatory positions related to a given regulatory issue.

TABLE 1. GLOSSARY

Clarify:	Revise text to make clearer the intent or implementation of the regulatory position
Convert:	Change text from one type of regulatory position into another type (e.g. Convert an Appendix of 10CFR50 into a Regulatory Guide) while maintaining the essential technical content
Eliminate:	Remove text
Expand:	Add text to increase the scope or implementation of the regulatory position
Regulation:	All text in Title 10 Code of Federal Regulations
Regulatory Position:	All text which describes regulatory requirements or methods which licensees and applicants for license may use to satisfy requirements (e.g. 10CFR, Standard Review Plan, Regulatory Guides, Generic Letters, industry standards)
Relocate:	Move text to enhance clarity and organization
Replace:	Substitute new text for existing text
Update:	Revise text to account for new information (e.g. referencable standards or research results)

3. RESULTS

Tables 2 through 4 summarize the recent results of this research. They list existing regulatory positions which are candidates for reexamination. The lists incorporate items held over from previous surveys[4,5a]. A more detailed description of the regulatory requirements and the associated potential actions follows the tables. The discussion groups related regulatory positions into the same unit for convenience.

Some general observations are noteworthy:

1. There are many specific existing regulatory requirements clearly identified as candidates for elimination or modification in the sense of the Commission's guidance. However, little success was achieved in pinning down the nature and extent of regulatory burden associated with a given regulatory position.

2. The NRC staff identified many additional specific items which represent updates, clarifications or expansions of current explicit regulatory authority.

3. There is a frequently expressed undertone of continuing concern about several general aspects of the regulatory requirements, including their currency, use of terminology, and initial operating conditions assumed for safety analysis.

4. It is apparent that formal rulemaking significantly lags the development and implementation of new regulatory positions in many cases. This introduces complexity and uncertainty in terms of exactly what requirements licensees of future reactors are expected to satisfy.

5. There is some sentiment toward cleansing 10CFR50 of all material not directly related to the safety of commercial nuclear power reactors and relocating it in some other part(s) of Title 10. The rationale is that 10CFR50 is complex enough for regulators and reactor licensees alike without having to deal with extraneous information such as requirements for medical therapy and research and development facilities, financial qualifications, and antitrust reviews. Such action would help reduce the clutter in the regulations (i.e., for power reactors) most in need of clarity.

TABLE 2. SPECIFIC CANDIDATES FOR CHANGE IN TITLE 10 OF THE CODE OF FEDERAL REGULATIONS

Potential Action	Regulatory Position	Subject
Eliminate/Clarify	Part 21 Part 50.9 50.55(e) 50.72 50.73	Reporting requirements
Eliminate	Part 50.13	Attacks and destructive acts
Convert/Relocate /Eliminate	Part 50.33(f)	Financial qualifications
Eliminate/Relocate	Part 50.33a	Antitrust review
Eliminate	Part 50.34(a)(4) (b)(4)	Contents of applications
Eliminate/Clarify	Part 50.34(f)	TMI-related requirements
Eliminate/Replace	Part 50.34(g)	Conformance with SRP
Convert	Part 50.44	Combustible gas control
Expand	Part 50.46 50 Appendix K	ECCS performance
Eliminate/Update	Part 50.49	Environmental qualification of electrical equipment
Update/Clarify/Expand	Part 50.55a	Codes and standards
Clarify	Part 50.59	Changes, tests and experiments
Replace	Part 50.62	ATWS requirements
Eliminate/Replace	Part 50.71(e)	FSAR updates
Replace/Expand	Part 50.73	LER system

TABLE 2. (CONTINUED)

Potential Action	Regulatory Position	Subject
Eliminate/Clarify	Part 50 Appendix A	Definitions and explanations
Update	Appendix A GDC2 Appendix A GDC4	Load combinations
Replace/Clarify/Update	Appendix A GDC19	Control room
Expand	Appendix A	Design against sabotage
Expand	Appendix A	General Operating Criteria
Expand	Part 50 Appendix B	QA for fraudulent products
Update	Part 50 Appendix B	Quality assurance in general
Convert/Relocate /Eliminate	Part 50 Appendix C	Financial qualifications
Update/Convert	Part 50 Appendix J	Containment leakage testing
Eliminate/Relocate	Part 50 Appendix L	Antitrust review
Convert/Update/Clarify	Part 50 Appendix R	Fire protection
Update/Convert /Eliminate	Part 100 Part 100 Appendix A	Reactor site criteria

TABLE 3. SPECIFIC CANDIDATES FOR CHANGE IN THE STANDARD REVIEW PLAN

Potential Action	Regulatory Position	Subject
Update	SRP 3.9.3	Load combinations
Eliminate/Update	SRP 3.11	Environmental qualification
Update/Convert	SRP 6.2.6	Containment leakage testing
Eliminate	SRP 6.7	Main steam isolation valve leakage control system
Update/Replace	SRP 13.2.1	Reactor operator training
Update/Expand	SRP 13.6 and related Div 5 RegGuides	Physical security
Update	SRP 14.2	Initial plant test program
Eliminate	SRP 15 sections e.g., 15.4.2 15.4.8 15.4.9 15.6.5B 15.6.5D	Design basis accident analysis Control rod withdrawal at power Rod ejection accidents (PWR) Rod drop accidents (BWR) Engineered safety feature leaks Main steam isolation valve leakage control system
Update	SRP 17.1	QA design and construction
Update	SRP 17.2	QA operations phase

TABLE 4. ADDITIONAL SPECIFIC CANDIDATES FOR CHANGE

Potential Action	Regulatory Position	Subject
Update	RegGuide 1.3 RegGuide 1.4	BWR source term for LOCA PWR source term for LOCA
Update	RegGuide 1.9	Diesel generator qualification
Expand	RegGuide 1.28	QA design and construction
Update	RegGuide 1.33	QA operations phase
Update	RegGuide 1.60	Seismic design response spectra
Update/Replace	RegGuide 1.61	Seismic design damping values
Update	RegGuide 1.76	Design basis tornado
Update	RegGuide 1.78	Control room habitability
Update	RegGuide 1.89	Environmental qualification of electrical equipment
Update	RegGuide 1.92	Seismic response analysis
Eliminate	RegGuide 1.96	BWR main steam isolation valve leakage control system
Replace	RegGuide 1.108	Diesel generator testing
Update	RegGuide 1.109	Calculation of radiological dose
Clarify	RegGuide 1.114	Control room boundaries
Eliminate	RegGuide 1.115	Turbine missiles
Expand	RegGuide 1.152	Digital computer software
Update	RegGuide 5.12	Security locks
Update	RegGuide 5.44	Perimeter intrusion alarms
Create/Expand	RegGuide	Credit for human performance
Eliminate	Generic Letter 82-28	Reactor vessel level indication

The information which follows describes in greater depth the specific candidates for change, the potential actions, and the nature and reasons for such actions. The order in which they are presented roughly parallels their presentation in Tables 2-4. However, regulatory positions which deal with similar regulatory issues are grouped for convenience.

Regulatory Position: 10CFR21 Reporting of defects and noncompliance
10CFR50.9 Completeness and accuracy of information
10CFR50.55(e) Conditions of construction permits
10CFR50.72 Immediate notification requirements
10CFR50.73 Licensee Event Report system

Potential Action: Eliminate/Clarify

Comments: Reporting requirements permeate 10CFR and are particularly prevelant in Part 50. Part 50.8 in fact specifies more than thirty sections of Part 50 which contain information requirements approved by the Office of Management and Budget under the Paperwork Reduction Act of 1980.

Among the requirements are those associated with reporting discovered defects. In issuing an Advanced Notice of Proposed Rulemaking (SECY-89-xxx), NRC is considering amending the regulations on the reporting of safety defects found during the design, construction, and operation of nuclear facilities. The proposed amendments would eliminate duplicate evaluation and reporting, establish a uniform threshold for defects that need to be reported and a uniform content for safety defect reporting, and establish consistent time limits for evaluation and reporting of defects.

Regulatory Position: 10CFR50.33(f) Contents of applications; general
Appendix C Financial qualifications for construction

Potential Action: Convert/Relocate/Eliminate

Comments: Part 50.33 specifies the general information an applicant for a construction permit or an operating licenses must submit as part of the application. It includes an unusual mix of basic data, such as the name, address and business of the applicant, as well as more expansive information such as financial qualifications and emergency response plans. Paragraph (f) specifically addresses "information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out...the activities for which the permit or license is sought."

Appendix C to Part 50, "A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Facility Construction Permits," amplifies on this requirement by citing more specifically what types and forms of information the Commission expects to receive. Both the regulation and the Appendix distinguish between applicants which are "established organizations" and those which "newly-formed" for the purpose of constructing or operating the licensed facility. The former are permitted to rely more on historical data such as financial statements; while the latter are expected to specify funding sources, assets, liabilities and the like. This same type of information would be required under 10CFR50.80 if an organization or individual wished to receive a license by transfer from another party.

An interesting aspect of Appendix C is its explicit expression as being a "guide" which is "not intended to be a rigid and absolute requirement." By common practice, such admission would more aptly define a Regulatory Guide than a regulation. Thus, one potential action is to convert Appendix C to a Regulatory Guide. Another potential action is to relocate these regulations to another part of 10CFR which is more procedurally oriented, perhaps combined with the antitrust review of 50.33a and Appendix L.

The more fundamental issue is the extent to which the required information is an appropriate indicator of the safe operation of the licensed facility. The extent to which a causal relationship between financial qualifications and safety can be defined and defended should dictate whether the elimination of these requirements should be considered.

Regulatory Position: 10CFR50.33a Information for antitrust review
10CFR50 Appendix L

Potential Action: Eliminate/Relocate

Comments: Section 105c of the Atomic Energy Act of 1954, as amended, directs the U. S. Attorney General to review antitrust aspects of the commercial nuclear power industry. The requirement for construction permit applicants to provide information for this review is specified in 10CFR50.33a. The information to be provided is specified in Appendix L. The purpose of the antitrust review is to assure that trade and commerce are protected from unlawful restraints and monopolies or unfair business practices.

The regulation divides applicants into groups based on their total electrical generating capacity: more than 1400 MW(e); 200 - 1400 MW(e); and less than 200 MW(e). The larger the applicant, the more information the applicant is required to provide. Further, the applicant must provide the required information as a separate document at least nine months but no more than 36 months in advance of any part of the application for construction permit. Separate documentation must be submitted for each application, regardless of prior similar submittals or reviews.

Appendix L describes 20 categories of information required in order to perform the antitrust review. Examples include data on loads and load growth, reserve capacity, alternative sources of generating capacity, transmission systems, neighboring utility systems, cost of power, corporate mergers with other electricity suppliers, and rates charged for power. A typical submittal for antitrust review is twenty copies of a 50-page (?) document.

The potential action is to eliminate this requirement from NRC's jurisdiction since it has no bearing on safety. An alternative action is to limit the required submittal to one time unless some significant change having antitrust implications has occurred since the last submittal. Finally another potential is to relocate the regulation and the Appendix out of Part 50, which is already complex enough, and into a more procedurally oriented part of Title 10.

It has been ten years since any organization applied for a construction permit. [How long? Who applied and for which plant?] During that time the electric generating industry has undergone significant changes, such as mergers of small generators into larger ones and the legislated requirement for large utilities to purchase power from small independent producers. The concept of companies whose sole function is to operate nuclear power plants (as opposed to transmitting and distributing electricity, for example) is receiving significant attention. Also the implementation of 10CFR52, an alternative process for licensing commercial nuclear power plants, is now in effect. Potential action on the antitrust review requirements would need to be viewed from these perspectives.

Regulatory Position: 10CFR50.34 Contents of applications; technical
(a)(4) Preliminary SAR
(b)(4) Final SAR
SRP 15 *Accident Analysis*; Multiple subsections
e.g., 15.4.8; 15.4.9; 15.6.5

Potential Action: Eliminate

Comments: FSARs must contain assessments which demonstrate that for certain postulated design basis accidents, the public would not be exposed to radiation doses in excess of 10CFR100 limits. Several related RegGuides provide further guidance on assumptions to be used in performing the calculations. The results (i.e. the calculated doses) are also used to establish equipment specifications for certain engineered safety features.

There are suggestions that many of the Chapter 15 calculations (e.g. rod drop and rod ejection events) usually have no meaningful risk significance, are costly to perform and review, and have little or no impact on plant design. In the few cases where dose criteria are calculated to be exceeded, exemptions are granted based on conservatisms in fuel failure assumptions. Thus the calculations are unproductive by any real measure and could be eliminated as a requirement.

The counterpoint is that the results of calculations for fuel handling accidents (SRP 15.7.4) and spent fuel cask drop accidents (SRP 15.7.5) can affect features of containment isolation systems. A different example is that for at least one case involving high burnup fuel, a calculated 20% increase in thyroid dose would cause some plants to exceed 10CFR100 limits. The required Chapter 15 calculations enabled this to become recognized. There is also the perception that licensees have well established analytical codes and procedures for performing Chapter 15 analyses and that the NRC staff spends little time reviewing the results. Thus the burden is low.

[Additional notes: 10CFR100.10 Factors to be considered when evaluating sites
10CFR100.11 Determination of exclusion area, low population zone, and population center distance
SRP 15.6.5 LOCA dose calculations
SRP 6.5.2 Containment Spray as a Fission Product Cleanup System
SRP 6.5.3 Fission Product Control Systems and Structures

Relative to using computed off-site doses to establish capacities for engineered safety features. This practice should be reconsidered since conservatisms in dose calculations overestimate doses which in turn lead to unnecessarily conservative designs. Consider replacing this practice, especially for new plants, with standard specifications (e.g.minimum capacities and efficiencies for engineered safety features) based on acceptable plant/site combinations, thus eliminating substantial unproductive effort. This would discourage compensating for a potentially disadvantageous site (as determined by dose calculations) by installing more and bigger engineered safety features. The new licensing process 10CFR52 offers the opportunity to modernize the approach.]

Regulatory Position: 10CFR50.34(f) Contents of applications; technical
Additional TMI-related requirements

Potential Action: Eliminate/Clarify

Comments: This regulation, commonly referred to as the "CP rule" was adopted in 1980 in the aftermath of TMI. It specifies requirements applicants must satisfy for an LWR construction permit or manufacturing license whose application was pending as of February 16, 1982. There are at least 50 major requirements imposed and most are quite specific. They include a plant/site specific PRA, various accident and reliability analyses, operability studies, improved simulation capability, improved operating procedures, control room design review, safety parameter displays, hydrogen control systems, valve qualification programs, QA program requirements, dedicated containment penetrations and many more.

The applicants to which this rule applies are mentioned by name in the text. As of January, 1989, all such plants have been cancelled. Thus, there is no immediate need for such a regulation. The continuing need for those requirements is based on the assumption that applications for new plants will be received and will have to comply with those requirements. Using the current text, it is unclear which specific requirements and design standards would apply to new plants. The process for licensing new plants is addressed in a new 10CFR Part 52, which should reference a modified 10CFR50.34(f) or other suitable regulations.

Regulatory Position: 10CFR50.34(g) Contents of applications; technical
Conformance with the Standard Review Plan

Potential Action: Eliminate/Replace

Comments: Part 50.34 Paragraph (g) requires all applications for operating licenses, construction permits, manufacturing licenses, and design approvals for standard plants docketed after May 17, 1892 (i.e., all future reactors) to contain "an evaluation of the facility against [the then current] Standard Review Plan." The regulation requires the applicant to identify, describe and justify any differences in design features, analytical techniques and procedures from those included in the acceptance criteria of the Standard Review Plan. It goes on to assert that "the SRP is not a substitute for the regulations, and compliance is not a requirement."

While the intent of the regulation is to enable flexibility without sacrificing assurance of safety, the realized effect is to discourage innovation. The disincentives arise from having to justify a departure from a previously accepted design feature or method, even though the designer and operator may believe the innovation represents a net improvement in safety, and the attendant uncertainty associated with its ultimate approval. The potential action would eliminate this part of the regulations or perhaps replace it with a more clearly explained statement of policy.

Regulatory Position: 10CFR50.44 Standards for combustible gas control system in light-water-cooled power reactors

Potential Action: Convert

Comments: This regulation, commonly referred to as the "hydrogen rule" was adopted in 1978 as a response to TMI and modified subsequently. It requires that every operating LWR be provided with means to manage combustible gas, primarily hydrogen, that can be generated during an accident. It sets standards for operating licenses and construction permits. The regulation is highly specific, comparable to those covering ECCS performance (50.46), emergency planning (50.47), fire protection (50.48), electrical equipment qualification (50.49), and their associated appendices. Its implementation has caused many changes in operating reactors including containment inerting, recombiners, high point vents, and reanalysis of containment response. A major issue here has been the regulation's assumption of 75% metal-water reaction and the implication that such extensive reactions can only occur beyond the plant's design basis.

The possible action is directed toward retaining the intent of the regulation but permitting licensees greater flexibility in satisfying the need. One possibility is to incorporate that intent into one or more of the General Design Criteria and relegating the more specific aspects of 50.44 to the Standard Review Plan or a RegGuide.

[For further information see Generic Letter 84-09 "Recombiner capability requirements of 10CFR50.44(c)(3)(ii)"]

Regulatory Position: 10CFR50.46 Acceptance criteria for ECCS
10CFR50 Appendix K ECCS Evaluation Models

Potential Action: Expand

Comments: As currently constituted, these regulatory positions address only fuels composed of uranium dioxide clad with Zircaloy. The potential action is to expand the scope of these and related regulatory positions by adding guidance for reload cores, some of which include cladding other than Zircaloy and some of which entail significantly higher burnup than was originally envisioned.

Regulatory Position: 10CFR50.49 Environmental qualification of electric equipment important to safety
SRP 3.11 Environmental Qualification of Mechanical and Electrical Equipment
RegGuide 1.89 Environmental Qualification of Certain Electrical Equipment Important to Safety

Potential Action: Eliminate/Update

Comments: This regulation deals with the ability of a broad array of equipment to survive the environments postulated to be associated with accidents in operating reactors. This rule was originally adopted in 1983, largely in response to findings of confirmatory research on the operability of electrical equipment in various thermal, radiation and humidity environments. The rule imposed a major burden on the licensees and NRC staff. It requires licensees to develop and execute an extensive program for qualifying their plants' electrical equipment, to document and report the results, to plan replacement of nonqualifying equipment, and to satisfy listed schedules for all of the above. At least three major controversies accompany this rule: the scope of equipment it covers, the environmental test conditions, and the implementation schedules.

The rule explicitly applies to "safety-related electrical equipment," "nonsafety-related electrical equipment" whose failure under the proposed environments would prevent satisfactory performance of the former, and "certain post-accident monitoring equipment." One potential action would be to reduce the scope of the rule to only that equipment whose malfunction directly impairs satisfactory achievement of a safety function, i.e. only safety-related electrical equipment. This reduction could be achieved by eliminating paragraphs (b)(2) and (b)(3) of the rule.

The rule specifies the environmental parameters (e.g., temperature, pressure, radiation, aging, etc) which must be included in the qualification program, as well as the need to consider synergistic effects and margins. RegGuide 1.89 is the more specific guidance on accident environments and test conditions. In essence it specifies environmental conditions attendant to design basis accidents, including radiological source terms characteristic of RegGuides 1.3 and 1.4. The potential action would be to modify the RegGuide to represent more realistically the environmental conditions suggested by more modern thought. Thus potential action on this matter is tied to potential action on the source term "megaissue."

All deadlines specified in the rule have passed. In some instances NRC has granted extensions for "good cause" or "sound reasons." Some licensees have still not demonstrated compliance. The rule contains no explicit provisions for applicability to operating licenses granted after November 30, 1985. Therefore it is unclear what requirements must be satisfied by future reactor designs.

Consideration of this issue sometimes leads to a related question regarding the environmental qualification of mechanical equipment, i.e. is electrical equipment overregulated or is mechanical equipment underregulated?

Regulatory Position: 10CFR50.55a Codes and standards

Potential Action: Update/Clarify/Expand

Comments: This Part was originally adopted in 1971 and has been revised nearly every year since then. It amplifies on General Design Criteria 36, 37, 39, and 40, which address inspection and testing of safety systems. The original focus of 50.55a was on requirements for In-Service Inspection (ISI). The regulatory philosophy was to endorse applicable current industry standards (e.g., ASME standards). It is one of the few (the only?) parts of the regulations which requires a licensee to modernize procedures and equipment in accordance with the latest approved industry standards.

In 1984, NRC modified the rule to include In-Service Testing (IST) primarily in response to the Davis-Besse loss of feedwater incident. The rule change required all 105 operating reactors to develop and document IST programs. As of late 1988, there were 12 such programs submitted, reviewed, and approved via a Safety Evaluation Report. The rest are in progress and are usually characterized by very many requests for exceptions.

The potential modification represents a package of changes which will collectively reduce regulatory burden and enhance safety. The more important specific changes suggested are:

- Segregate the text addressing IST from that addressing ISI and add supplemental clarifying text where appropriate
- Eliminate the requirement for NRC to review and approve every change to a licensee's ISI and IST programs
- Encourage (require?) a process whereby deviations from ASME Codes are justified by licensees rather than preapproved by NRC
- Develop a Reg Guide which documents generally approved deviations from the ASME Code (e.g., Section XI incorporating OM-6 and OM-10)
- Require licensees to maintain all relevant information on-site and available for inspection
- Place more emphasis on system performance relative to component testing
- Impose a one-time mandatory update to the ISI and IST programs of all operating reactors in the 1992 time frame; then allow all subsequent updates to be voluntary except for ASME Code revisions
- Develop/introduce standards and criteria for testing the instrumentation and control portion of safety systems
- Permit more flexibility in scheduling "ten-year inspections" at multiple unit sites.

The potential actions are motivated by a need to improve the in-service programs themselves as well as the process by which they are reviewed and implemented. The suggested changes would represent a major impact on licensees with relatively weak in-service programs and a significantly lesser impact on licensees that currently have stronger programs.

[For further information see Memorandum Murley to Beckjord, "Proposed amendment to 10CFR50.55a - Codes and Standards," May 12,1988.]

Regulatory Position: 10CFR50.55a(h) Codes and standards
Protection systems

Potential Action: Update

Comments: The paragraph currently codifies IEEE 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations." An industry working group is currently revising IEEE Std 279-1971.

The rule should be updated to codify IEEE Std 603-1980 "Criteria for Safety Systems for Nuclear Power Generating Stations," a newer, more comprehensive standard. The requirements and recommendations of IEEE Std 603-1980 on the power, instrumentation and control portions of safety systems incorporate the requirements and recommendations of IEEE Std 279-1971, which is limited to protection systems only. Currently operating reactors should be grandfathered.

Regulatory Position: 10CFR50.59 Changes, tests and experiments

Potential Action: Clarify

Comments: Under Part 50.59, a licensee may make a change, test or experiment without prior approval by the NRC unless the change, test or experiment involves a change in a technical specification or an "unreviewed safety question." The rule then proceeds to define what the licensee must do in the latter case in terms of recordkeeping, safety analysis, and reporting.

The crux of this matter is in determining if a change involves an unreviewed safety issue, the so called "50.59 evaluation." The rule provides descriptive language to help define an unreviewed safety issue, the basic intent of which is to assure that the plant remains within the envelop represented by the safety analysis report. The evaluation focuses on design basis events only, since these are the events of interest in the plant's safety analysis report. Further, if the licensee performs such an evaluation and concludes that a technical specification change or an unreviewed safety issue is involved, such change, test or experiment has to be achieved through an amendment to the operating license pursuant to Part 50.90, a substantially involved process which licensees usually seek at avoid.

The potential action is to clarify 50.59. Most commenters understand the need for the regulation, but many find its language ambiguous, vague or confusing. The result is perceived to be too many evaluations submitted which are not really significant from a safety standpoint and probably some evaluations which are significant but are not submitted for review by the NRC. An industry task force (NUMARC?) is currently reviewing this matter at the request of NRC's management (5/27/86). Their task is to develop review criteria and guidelines for licensees conducting 50.59 evaluations. They expect to issue their findings and recommendations in (when?).

[For further information see Bryan or Butcher.]

Regulatory Position: 10CFR50.62 Requirements for the reduction of risk from ATWS

Potential Action: Replace

Comments: After many years of technical analysis and debate, the NRC adopted Part 50.62 in its final form in 1984. The regulation applies to all commercial light water reactors. Within it there are specific additional elements which apply to particular types of reactors only. Boiling water reactors receive particular attention. The requirements relate to hardware "fixes" and include a schedule for final implementation.

Once the currently operating reactors have modified to comply with the regulation, the only continuing purpose for the regulation is enforcement. Some suggest that for future reactors, ATWS will be "designed away" and thus can be accommodated within the General Design Criteria. The potential action is to limit the current 50.62 to currently operating reactors and for future reactor designs to replace it with some as yet unspecified alternative documented position.

Regulatory Position: 10CFR50.71(e) Maintenance of records, making of reports Updates to the FSAR

Potential Action: Eliminate/Replace

Comments: Part 50.71 addresses the retention of records and reports required by the regulations, technical specifications and conditions of the license. Paragraph (e) of this part specifies further that all operating licensees must periodically update the Final Safety Analysis Report and submit such updated information to the NRC. Examples of required updates includes the effects of all physical or procedural changes to the facility as described in the FSAR, all safety evaluations supporting license amendments or Part 50.59 evaluations, and all analyses of new safety issues. The rule provides a timetable for updating FSARs and adds an information item specific to operating reactors which were included in the NRC's Systematic Evaluation Program (SEP) at the time.

Paragraph (e) became effective in July 1980 in response to difficulties the NRC staff was encountering in assuring current information on the design and operation of plants needed to evaluate safety. It was also a time when significant changes were being implemented in response to post-TMI requirements, and there was a perceived need to assure that such wholesale changes in the plants were recorded in an orderly manner.

There are several potential actions possible. The first is to eliminate that text which distinguishes among plants that were and were not in the SEP. That language has outlived its usefulness. The second is to maintain the requirement for regularly updating the FSAR but replace that portion which makes submittal of such updates to the NRC mandatory. The licensee could be required to maintain and produce on demand a current FSAR at the plant site or other similarly appropriate location. The final option is to eliminate Paragraph (e) all together on the grounds that changing the FSAR has no direct impact on the plant's safety and represents only a burdensome exercise in generating and moving paper.

Regulatory Position: 10CFR50.73 Licensee event report system

Potential Action: Replace/Expand

Comments: This part defines reportable events and reporting procedures. The broad issue this part has traditionally raised is the scope of reportable events and whether such reports should be required versus voluntary. A specific instance has been raised wherein paragraph (a)(2)(vi) would allow the failure of an important component, such as an emergency diesel generator, to remain unreported by a licensee if a redundant component performed successfully. Left unreported, this could lead to an underestimation in the rate of diesel generator failures at a time when the use of failure rate data for PRAs is increasing. It could also allow a series of seemingly unrelated failures at many sites to continue unattended.

If they so elect, licensees may report such failures to INPO's NPRDS or maintain their own records. The potential action is to delete 10CFR 50.73(a)(2)(vi) and introduce clarifying or replacement language which adds reporting requirements for important components. An alternative is to require licensees to keep plant-specific records of such failures on-site without having to report them to NRC.

[For further information see Generic Letter 83-43 "Reporting requirements of 10CFR50, Sections 50.72 and 50.73 and STS."]

Regulatory Position: 10CFR50 Appendix A General Design Criteria
Definitions and Explanations

Potential Action: Eliminate/Clarify

Comments: Adopted in 1971, this section contains two footnotes intended at the time to provide clarifying information. Footnote 1 addresses loss of coolant accidents and says "Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development." There appears to be no continuing need for this footnote, so it can be eliminated.

Footnote 2 is intended to clarify the definition of single failure. The definition refers to active and passive components of fluid and electrical systems. The footnote says "Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development." Traditionally, in electrical systems no distinction is made between active (e.g. switches) and passive (e.g. wires) components in terms of system failure, thus the footnote does nothing to amplify the basic definition. There also appears to be no apparent need for the reference to passive components in fluid systems.

Regulatory Position: 10CFR50 Appendix A GDC 2 Design Basis for Protection Against Natural Phenomena
GDC 4 Environmental and Dynamic Effects Design Bases
SRP 3.9.3 ASME Code Class 1, 2, and 3 Components, Component Support, and Core Support Structures

Potential Action: Update

Comments: One or more of the above should be updated to allow decoupling of design basis accident (e.g. pipe rupture) loads from seismic loads as they pertain to the design of mechanical equipment. Revisions are justified by the clearly distinguishable timing and duration of such loads which are now required to be combined. Such revisions would apply only to mechanical equipment and would primarily affect BWR's. (Most PWR's can qualify for leak-before-break and most BWR's do not.) Some staff believe that even for some piping runs in PWR's (such as main steam lines), leak-before-break does not apply and the existing regulatory positions should be retained.

Updating SRP 3.9.3 has ramifications for SRP 3.8.1 through 3.8.4, which address structural aspects (including load combinations) for containments. Any of the above revisions, if made, should retain the coupling of accident and external event loads for structures.

[For further information see Memorandum Arlotto to Speis, "Recommended actions regarding decoupling of seismic and pipe rupture loads," June 9, 1986, and Memorandum Scinto to Arlotto, "SRP 3.9.3," May 15,1986.]

Regulatory Position: 10CFR50 Appendix A GDC 19 Control room
RegGuide 1.78 Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release
RegGuide 1.114 Guidance to Operators in the Control Room

Potential Action: Replace/Clarify/Update

Comments: GDC 19 requires licensees to provide a control room from which actions can be taken to operate the plant safely and to maintain it in a safe condition under accident conditions. It also requires the room be adequately protected from radiation such that personnel in the control room during an accident receive no more than 5 rem whole body or its equivalent for the duration of the accident. This dose limit often is the determining factor driving design considerations for control room habitability.

One potential action is to replace the current 5 rem limit with a new limit of 25 rem. The rationale is that emergency workers are permitted to receive up to 25 rem during an accident, and that during an accident control room staff would be considered emergency workers. [From where does the alleged 25 rem dose limit derive?]

Another potential action is to review and update the current regulatory positions associated with habitability of control rooms, particularly in reference to hazardous chemical releases. The current RegGuide 1.78 was issued in 1974 and has never been revised.

A related potential action is to clarify what constitutes the physical boundaries of the control room as that term is used in 10CFR50.54(m). Such action is underway through a revision of RegGuide 1.114 in progress.

Regulatory Position: 10CFR50 Appendix A General Design Criteria
10CFR50.13

Potential Action: Expand/Eliminate

Comments: As currently formulated, the General Design Criteria contain no specific provisions for designing against sabotage. Recent reviews of new standard designs rely on NUREG-0908 "Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans" August 1982 for regulatory guidance. The suggested action is to synthesize into a new General Design Criterion the accumulated wisdom gathered from past experience.

Interestingly, 10CFR50.13 explicitly advises applicants that they are "not required to provide for design features or other measures for the specific purpose of protection against the effects of (a) attacks and destructive acts, including sabotage, directed against the facility by an enemy of the United States, whether a foreign government or person, or (b) use or deployment of weapons incident to U. S. defense activities." Adopted in 1967, the language could be interpreted to apply only or primarily to acts of war. A more rigorous interpretation could suggest that sabotage need not be a design consideration in any sense. This part would imply conflict with the potential expansion of Appendix A, such that any inconsistency between 10CFR50.13 and an expanded Appendix A would need to be resolved. Part 50.13 also exemplifies a common instance where the text of the regulations is used to define what is not required, rather than what is required.

Regulatory Position: 10CFR50 Appendix A General Design Criteria

Potential Action: Expand

Comments: The General Design Criteria were originally adopted in 1971 to "establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission." Despite explicit expectations of change, the General Design Criteria have remained relatively stable over the years since their adoption, though the voluminous subsidiary implementing regulatory positions have swelled.

The potential action is to develop a comparable set of "General Operating Criteria" to help regulate plant operations analogous to the way the GDC establish basic principles for plant design. The focus of reactor regulation has shifted from design/construction in the sixties and seventies to operations in the eighties and beyond. It is apparent that the regulations have not kept pace with this shift. One example of a "deficiency" is that nowhere do the regulations require an ongoing training program for plant maintenance staff. Nor have requirements for emergency procedures, control room shift staffing, and safety parameter display systems, which were implemented via Generic Letters after TMI, ever been integrated effectively into the regulations.

There had been an attempt to develop a few such GOC soon after TMI, but the effort withered. The Institute of Nuclear Power Operations has developed criteria to guide their evaluations of the nuclear reactor industry, but under the current institutional relationships, those criteria are voluntary.

Regulatory Position: 10CFR50 Appendix B Quality Assurance Criteria
RegGuide 1.28 Quality Assurance Program Requirements
(Design and Construction)

Potential Action: Expand

Comments: Appendix B and RegGuide 1.28 establish quality assurance controls for procuring products and services. Both positions enable licensees to enhance their ability to detect defective workmanship, but neither were specifically developed with fraudulent products in mind. Recent discoveries of such products in operating reactors leads to the recommendation for modifying the regulations to lessen the associated risk.

The action proposed is to "strengthen" Appendix B. NRC has issued Generic Letter 89-02, which addresses this matter in part by conditionally endorsing industry efforts (EPRI NP-5652 "Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications"). The letter describes modifications to the EPRI Guidelines which will "satisfy existing requirements of Appendix B" and "reduce the likelihood of the introduction of counterfeit or fraudulent products." The staff has also issued an Advanced Notice of Proposed Rulemaking (SECY-89-010) soliciting public comment on the matter.

Regulatory Position: 10 CFR 50 Appendix B Quality Assurance Criteria
SRP 14.2 Initial Plant Test Program FSAR
SRP 17.1 Quality Assurance Design and Construction
SRP 17.2 Quality Assurance Operations Phase
RegGuide 1.28 Quality Assurance Program Requirements Design and Construction
RegGuide 1.33 Quality Assurance Program Requirements Operations Phase

Potential Action: Update

Comments: Appendix B contains the basic requirements for quality assurance. Beyond Appendix B are many different versions and interpretations of the same basic requirements. Contributing to the confusion is the increased tendency to deal with immediate safety concerns through the mechanism of Generic Letter and then failing to follow through in timely fashion with revisions to appropriate regulations, SRP sections, and RegGuides.

A case in point is the matter of post-trip review. Generic Letter 83-28 identified required actions based on the generic implications of the Salem ATWS event. Two subsequent Generic Letters (85-09 and 85-10) added additional technical specifications. Yet the relevant SRP Section 4.3 Nuclear Design and RegGuide still contain no such requirements. Nor do recent industry standards (e.g., ANS 3.2-1988) appear to be current.

The potential action is to make a concerted effort to review and update the quality assurance aspects of all current regulatory positions. This would benefit current licensees and future reactors.

Regulatory Position: 10CFR50 Appendix J Primary Reactor Containment Leakage Testing
SRP 6.2.6 Containment Leakage Testing

Potential Action: Update/Convert

Comments: Appendix J was originally adopted in 1973 to support 10CFR50.54(o) and General Design Criteria 50 through 57. Licensees must demonstrate that their plants satisfy the containment leakage testing requirements of Appendix J as a condition of the operating license. Three types of tests are defined:

- Type A - an integrated leak test at not less than design basis pressure
- Type B - tests for local leaks around containment penetrations
- Type C - tests for local leaks in containment isolation systems.

The Appendix codifies the relevant ANSI Standard N45.4-1972, specifies other acceptable test methods, sets requirements for scheduling tests, specifies test parameters, establishes acceptance criteria, and defines procedures for validating and reporting test results.

The principal thrust of the potential actions is to bring practices for containment leak testing more in line with recent insights from probabilistic risk assessments. A secondary consideration is that the significant amount of detail in this Appendix makes it better suited as a RegGuide than a regulation. There is also continuing concern within the industry about the significant costs of containment leak testing.

Appendix J has been recognized as a candidate for major revision since results of early risk assessments showed risk to be dominated by accidents involving core damage and major breaches of containment (as opposed to the relatively intact containment integrity typified by Appendix J). A prior detailed review of this subject (NUREG/CR-4330) done according to NRC value-impact guidelines concluded that a 100-fold increase in allowable leak rates could be permitted without significant adverse effects on public safety. No rule change resulted from this finding. The NRC did modify Appendix J in 1988 to permit use of the Mass Point method of leak detection, a practice which had previously required exemptions. Still the major issues of test type, test frequency, and allowable leak rates remain unchanged.

Regulatory Position: 10CFR50 Appendix R *Fire protection program*

Potential Action: Convert/Update

Comments: General Design Criterion 3 *Fire protection* requires licensees to design and locate structures, systems and components important to safety to minimize effects of fires and explosions and to provide appropriate systems for detecting and fighting fires. In 10CFR50.48 the NRC requires licensees to have a fire protection plan which expands on the physical considerations of GDC 3 and adds administrative and personnel considerations. That part proceeds to reference Appendix R, set schedules for compliance, and establish procedures for review and approval.

Adopted originally in November 1980 as a response to the Brown's Ferry fire, Appendix R "establishes fire protection features required to satisfy" GDC 3. The Appendix consists of nearly eight pages of "general requirements" and "specific requirements" related to hazards analysis, equipment, barriers, safe shutdown capability, water supplies, detection, fire brigade, training, records, and other related matters. The NRC staff has issued four related Generic Letters:

81-12	Safe shutdown capability after fires
83-33	NRC positions on Appendix R
86-10	Implementation of fire protection requirements
88-12	Move fire protection program from technical specifications to FSAR.

There was an apparent lack of referencable industry standards when Appendix R was being developed. That situation appears to persist today.

The highly detailed nature of these requirements leads some to suggest that Appendix R would be more appropriate as a RegGuide than a regulation. Most existing reactors have complied or received exemptions. Much of Appendix R has been incorporated into SRP 9.5.1, so duplication exists. Retaining Appendix R in 10CFR50 assures the regulatory staff of a continuing source of enforcement authority.

New plants are expected to comply with Appendix R without exception, though the regulation does not mention this explicitly. There is some concern that for new plants Appendix R is necessary but insufficient, thus meriting some additional modification (e.g. increased separation distances). Resolution of this matter is achieved during the review process for new plants. ?

Regulatory Position: 10CFR50 Appendix R Fire Protection
Paragraphs III.L.3-7 Alternative and dedicated shutdown capability

Potential Action: Clarify

Comments: These paragraphs relate primarily to the need for on-site vs. off-site power to electrify equipment for cold shutdown. (More to follow)

Regulatory Position: 10CFR100 Reactor Site Criteria
10CFR100 Appendix A Seismic and Geologic Siting Criteria for Nuclear Power Plants

Potential Action: Update/Convert/Eliminate

Comments: Part 100 constitutes the basic requirement regarding factors important in determining site suitability for power reactors. Appendix A contains more detailed requirements on the seismic and geologic aspects of siting. These regulations were adopted "as an interim guide" in 1962 and have remained essentially unchanged since the 1970's. Substantive updates are proposed in two areas; radiological source terms and geological science.

The thrust of the potential action is to use more effectively the probabilistic insights gained in the last decade in time to benefit the licensing reviews of new reactors. Possible modifications related to the radiological source term would affect Part 100.11, which deals with determining exclusion area, low population zone, and population center distance. One suggestion is to eliminate the footnote referencing Technical Information Document 14844 (March 23, 1962) for "further guidance". Doing so would make more apparent that alternatives to strict compliance with TID 14844 may be acceptable. The effects such elimination would have on subsidiary regulatory positions (e.g. Regulatory Guides 1.3 and 1.4) would need to be evaluated carefully.

The second area for modification is Appendix A. Technical updates would address the definitions of the Safe Shutdown Earthquake and the Operating Basis Earthquake as well as the relationship between the two. The SSE is defined by the response spectra corresponding to the maximum vibratory ground motion associated with faults in the vicinity. Certain safety systems must remain functional during SSE. The OBE is that earthquake producing "vibratory growth (sic) motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional." Traditionally the SSE is twice the amplitude of the OBE. While this traditional approach generates a conservative stance regarding safety, some believe that the associated OBE is unnecessarily conservative and engenders significant cost burdens in terms of structures and equipment. One suggestion is to allow the applicant to determine the OBE, thus increasing flexibility, while the NRC staff retains authority to determine the SSE. An alternative approach is to convert Appendix A into a RegGuide, thus theoretically introducing more flexibility into the process.

A significant overhaul of Part 100 has major implications in terms of staff effort, industry effort, and reactor licensing. There are many subsidiary regulatory positions potentially affected by changes to Part 100. Some believe the long term benefits are important enough to merit priority over revisions to Part 50.

Regulatory Position: SRP 6.7 Main Steam Isolation Valve Leakage Control System (BWR)
SRP 15.6.5D Radiological Consequences of Design Basis LOCA: Leakage from Main Steam Isolation Valve Leakage Control System (BWR)
RegGuide 1.96 Design of Main Steam Isolation Valve Leakage Control Systems for BWR

Potential Action: Eliminate

Comments: General Design Criterion 54 "Piping Systems Penetrating Containment" requires, in part, that piping systems penetrating containment be provided with leak detection, isolation and containment capabilities having redundancy, reliability and performance capabilities that reflect the importance to safety of isolating these piping systems. Operating experience in the early 1970's showed degradation of BWR MSIV's. This led to supplemental design features to control and contain the leakage of radioactive material from MSIV's as described in RegGuide 1.96 and SRP 6.7. SRP 15.6.5D describes acceptable means for calculating the release of fission products and their contribution to off-site doses following a large break LOCA.

The principal thrust of the potential action is to bring practices and designs related to MSIVLCS's more in line with recent insights from probabilistic risk assessments. There is a range of potential elimination as well. One could opt to eliminate only the Chapter 15 calculations on radiological consequences. More aggressively, one could opt to eliminate aspects of the design, inspection or testing of the MSIVLCS. One possibility is to downgrade MSIVLCS to a nonsafety system. In the extreme, one could opt to eliminate or shut down totally the MSIVLCS.

This subject has been recognized as a candidate for elimination for several years. A prior detailed review of the matter (NUREG/CR-4330) using NRC value-impact guidelines concluded that, if treated as a new requirement for operating reactors, the MSIVLCS would not be justified as a backfit. Going further, the elimination of MSIVLCS would generate substantial savings without significantly increasing risk. Despite this evidence the NRC staff has held its position on MSIVLCS, in part because there are dose calculations for at least one plant (Nine Mile Point) for which accidents involving MSIV leakage was a limiting case.

Regulatory Position: SRP 13.2.1 Reactor Operator Training

Potential Action: Update/Replace

Comments: SRP 13.2.1 was originally issued in July 1981 as one mechanism for codifying the TMI Action plan requirements related to reactor operators. It specifies what the Preliminary and Final Safety Analysis Reports must contain regarding training programs for reactor operators and senior reactor operators in order for an applicant to receive a license. The Section references the relevant parts of the regulations (10CFR Part 50.54 (i) - (m); Part 50 Appendix A; Part 50.55); RegGuides (1.8, 1.149); and other regulatory positions (NUREG-0094, NUREG-0718, and NUREG-0737).

The potential action is to update SRP 13.2.1 to reflect more current regulatory guidance. One particular action mentioned frequently is to replace NUREG-0094 "NRC Operator Licensing Guide" with NUREG-1021 "Operator Licensing Examiner Standards" October 1983. Referenced RegGuides have been revised (e.g., RegGuide 1.8 Qualifications and Training of Personnel for Nuclear Power Plants endorses ANSI/ANS 3.1-1981) but remain to be incorporated into the SRP. An updated SRP 13.2.1 should also reflect the Commission's 1985 and 1988 policy statements on training as well as the current 10CFR55. For example, in at least four places, SRP 13.2.1 references 10CFR55.22, a paragraph which does not exist in Part 55.

Another point of view suggests that any training program accredited by the Institute of Nuclear Power Operations could serve as an adequate replacement for the specific positions defined in SRP 13.2.1.

[For further information see K. Perkins and the following Generic Letters:

88-13 Operator licensing exams
88-09 Pilot testing of fundamentals exam
87-14 Operator licensing exams
87-07 Transmittal of final rulemaking for revisions to operator licensing - 10CFR55 and conforming amendments

Regulatory Position: SRP 13.6 Physical Security
All Division 5 RegGuides related to security, e.g.,
RegGuide 5.12 Use of locks for protection and control
RegGuide 5.44 Perimeter intrusion alarm systems

Potential Action: Update/Expand

Comments: All existing RegGuides on security have been rendered obsolete by newer security technology. They do not apply to such advances as microprocessor controlled security systems and thus would be of little value in licensing new replacement equipment for operating reactors or for new plants.

NUREG-0908 "Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans," August 1982 presents a more modern perspective and is current practice for regulatory reviews, but is still not current with modern security technology.

Regulatory Position: SRP 16.0 Technical Specifications
NUREG-xxxx Standard Technical Specifications

Potential Action: Expand/Clarify

Comments: Current requirements do not clearly address exceedance of the Operating Basis Earthquake. Matters such as what defines the exceedance of the OBE, the associated reporting requirements, and the criteria for restart are not codified except by precedence. Industry working groups are preparing draft positions on the matter. NRR is also considering a new focused rule addressing instrumentation for seismically-induced shutdown.

[For further information see Memorandum Treby to Bagchi, "Interpretation of Part 100, Appendix A regarding: Proposed guidelines for determining when Operating Basis Earthquake is exceeded," May 3, 1988.]

Regulatory Position: RegGuide 1.3 BWR source terms for LOCA analysis
RegGuide 1.4 PWR source terms for LOCA analysis

Potential Action: Update

Comments: RegGuides 1.3 and 1.4 provide acceptable assumptions for use in calculating potential radiological consequences from postulated loss of coolant accidents in BWRs and PWRs respectively. As two of the oldest RegGuides, their basic assumptions about radiological source terms have influenced many subsequent regulatory positions. The basic perceived problem with these positions is that they overstate the release of radioactive material during an accident and as a result contribute to unnecessarily conservative design.

One example of this is Paragraph C.1.f of RegGuide 1.3 which states in its entirety "No credit shall be given for retention of iodine in the suppression pool." Based on research results, the NRC staff has already modified Sections [which ones?] of the Standard Review Plan to permit credit. The potential action would extend this modification to the RegGuide. Other examples of behavior in which research has changed perceptions of conservatism involve the fate of iodine isotopes, the nature of nonvolatile radioactive species, and the distribution of energy sources.

Regulatory Position: RegGuide 1.60 Seismic Design Response Spectra

Potential Action: Update

Comments: Update to reflect newer data on Eastern earthquakes. (More to follow)

Regulatory Position: RegGuide 1.61 Seismic Design Damping Values
RegGuide 1.92 Seismic Response Analysis Modal Responses

Potential Action: Update/Replace

Comments: Endorse ASME Code Case N411 and resolution of Unresolved Safety Issue A-40. This was done in RegGuide 1.84 specifically for piping. (more to follow)

Regulatory Position: RegGuide 1.76 Design Basis Tornado

Potential Action: Update

Comments: Revise to reflect modern knowledge. Could relax or endorse national standard (more to follow).

Regulatory Position: RegGuide 1.92 Seismic Response Analysis Modal Responses
RegGuide 1.122 Seismic Response Spectra Floor Design

Potential Action: Update

Comments: These RegGuides address acceptable methods for calculating the stiffness of concrete structures other than containment. Research results generated recently at Los Alamos and Taiwan suggest that the current methods may overestimate stiffness and thus should be considered for revision.

Regulatory Position: RegGuide 1.108 Emergency Diesel Generator Periodic Testing
RegGuide 1.9 Emergency Diesel Generator Qualification

Potential Action: Replace/Update

Comments: RegGuide 1.108 specifies testing for emergency diesel generators and includes provisions for cold start. The cold start is expected to confirm reliable start of the emergency power supply in the event of a large break LOCA (?) coincident with loss of normal power supply. Unfortunately cold starts increase the wear and tear on diesels and could actually decrease their reliability over the long term. Furthermore, the significance of large break LOCA as an initiating event is less than originally believed (relative to accidents which develop more slowly).

The suggested action is to withdraw RegGuide 1.108 and incorporate updated guidance into RegGuide 1.9. The updated guidance would reduce the frequency of cold start test to once per six months and would permit some warm start-up as expressed in Generic Letter 85-14. This action is in progress along with several related changes/updates on calculating loss of AC power duration and diesel generator reliability (RegGuide 1.155 Station Blackout).

Regulatory Position: RegGuide 1.109 Calculation of Annual Doses to Man

Potential Action: Update

Comments: RegGuide 1.109 provides the equations by which the NRC staff estimates radiation exposures for maximum individuals and population within 50 miles of the plant site. These equations yield the dose rates to various organs from various exposure pathways. Revised last in 1977, it is used primarily to demonstrate compliance with 10CFR50 Appendix I (ALARA). However, it also serves to provide acceptable methods for calculating doses to control room staff during postulated accidents (SRP 6.4) and off-site doses for various postulated accidents (RegGuides 1.3, 1.4, 1.25, and Safety Guide 5.)

The potential action would update RegGuide, particularly the dose conversion factors and whole body/organ dose equivalents, with current information from ICRP 26. [Such revision is either already underway or is scheduled to follow current revisions to 10CFR20. See H. Petersen.]

Regulatory Position: RegGuide 1.115 Protection against Low-Trajectory Turbine Missiles

Potential Action: Eliminate

Comments: General Design Criterion 4 "Environmental and Missile Design Bases" protection of structures, systems and components important to safety against the effects of missiles that might result from equipment failures. The "equipment failures" of principal interest at the time were overspeed failures of main turbine-generator sets. The operating experience available at the time suggested that protection of important components against missiles from turbine failure was an appropriate safety consideration, particularly since many early plants had turbines oriented tangentially to the containment.

RegGuide 1.115 describes acceptable methods for showing that the risk from turbine missiles is acceptably small, either through spatial orientation or physical protection. The RegGuide was last revised in 1977. Since then, newer plants have been designed with the turbines oriented radially to the containment. In addition there have been substantial improvements in turbine materials, turbine monitoring and overspeed protection which appear to have substantially reduced the risk of catastrophic failure.

The potential action is to eliminate RegGuide 1.115. The NRC staff no longer uses it, preferring instead to focus on the procedures and schedules for turbine inspection. On the other hand, the RegGuide imposes no apparent significant burden on anyone and remains sound design guidance.

Regulatory Position: RegGuide 1.152 Criteria for Digital Computer Software

Potential Action: Expand

Comments: RegGuide 1.152 describes acceptable methods for complying with GDC 21 "Protection system reliability and testability" as applied to safety related systems using programmable digital computer systems. The method applies to designing software, verifying software, implementing software, and validating computer systems. The RegGuide endorses ANSI/IEEE-ANS-7-4.3.2-1982 "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations" as a method acceptable for designing software, verifying software, implementing software, and validating computer systems used in safety-related systems. That standard complements several others which also relate to safety systems:

IEEE Std 279-1971 Criteria for Protection Systems
IEEE Std 467-1980 [Title?]
IEEE Std 603-1980 Standard Criteria for Safety Systems (hardware)
ANSI/ASME NQA-1-1979 [Title]

The nature (the existence?) of an issue here is somewhat unclear. In the late 1970's when a digital protection system was proposed for Arkansas Nuclear One, there was considerable consternation and effort to achieve reasonable assurance that the system would perform satisfactorily. Concerned about an error in the software, the NRC staff performed a line-by-line review. Though the system was ultimately approved, the experience deterred both licensees and staff from encouraging submittal of digitally-based designs.

In the interim, computer technology (e.g., microprocessors, fiber optics, PC's, etc) has progressed exuberantly in general and in safety-related applications in other industries. New standards have emerged. The advantages offered by digital systems over analog in terms of accuracy, reliability, versatility and cost have become generally recognized. On the other hand, design errors or failures in digital systems may be more subtle or difficult to detect. Yet there are no specific requirements and few documented regulatory positions to guide designers, licensees and reviewers. For example, NUREG-0700 "Review Criteria for Control Room Design Evaluations" does not meaningfully address new digital instrumentation, control and displays.

This is likely to remain an issue as licensees seek to refurbish operating reactors and as new designs are submitted for design certification. Current practice appears to sidestep the issue until forced by a particular event. For certification of new designs, the procedure appears to involve a "licensing basis agreement" with designers which commits to the use of best available industry standards [L. Rubenstein?].

Regulatory Position: None existing;
Possible RegGuide on credit for human performance

Potential Action: Expand/Create

Comments: The evolution of reactor designs has been such that designers have placed specific reliance on the performance of reactor operators to terminate design basis events. Where such reliance exists, design guidance in the form of a current standard "20-minute rule" has been used frequently.

With the advent of more investigation into events beyond design basis, there is a growing tendency to claim that plant operators and other plant personnel are capable and will take action necessary to manage an accident. What usually ensues is substantial discussion about why or why not such claims are valid.

The potential action would create a RegGuide or expand an existing one which provides guidance to licensees on an acceptable method for justifying credit for a specified operator action. An example is the action necessary to open a wetwell vent line manually during a station blackout sequence in a boiling water reactor. In order to receive credit for such action, the licensee could show the following:

- an analysis of the physical behavior of the plant under such conditions
- a procedure describing the specified action, including any tools or other assistance or personnel the procedure may entail
- certification that relevant training has been provided to those who must implement the procedure.

Regulatory Position: Generic Letter 82-28
Reactor Vessel Level Indication System

Potential Action: Eliminate

Comments: The requirement for a means of unambiguously determining the water level in the reactor pressure vessel was imposed via generic letter as TMI Action Plan Item (NUREG-0730 II.F.2). The rationale was that had such an indication been apparent to the control room crew at TMI, they would have acted to restore inventory and flow through the core in the critical early hours of the accident.

To date, [number] reactors have in them a Reactor Vessel Level Indication System. For boiling water reactors [some description of status.] For pressurized water reactors [some description of status.] Most licensees characterize the RVLIS as a required system whose benefit is unproven and thus whose cost is unjustified. Licensees propose that newly revised emergency operating procedures in use by trained operators assure identification of threats to core uncovery and thus satisfy the same safety function as would a backfitted RVLIS at considerably less cost. In those systems where RVLIS has been installed, operating performance and reliability has been "poor."

The potential action is to delete the requirement for RVLIS as it applies to currently operating reactors. For future reactors, it appears to be more practical to design a means for directly measuring water level in the pressure vessel. Incorporating RVLIS into the design should also lead to improved performance and reliability versus backfitted systems. Thus this requirement could be retained on the basis of lower cost for similar benefit.

[Despite repeated opportunities, no one on the staff defended RVLIS, but no one proposed to eliminate it either. Its inclusion here is primarily a carry-over from the 1985 review of regulations. A similar situation existed for Safety Parameter Display Systems.]

Regulatory Position: General - Initial conditions

Potential Action: Expand

Comments: For the most part, the regulations and associated regulatory positions assume the starting point for safety analysis to be operation at full power, an equilibrium condition, and all safety systems in compliance with technical specifications. In some instances, licensees have been required to perform safety evaluations from other conditions, such as 5% power. Operating experience and analyses [reference?] suggest that there are other operating modes in which the possibility of serious consequences may not be fully appreciated. For example, during maintenance outages, loss of shutdown heat removal could lead to core damage with safety systems out of service and a relatively open containment.

The potential action is to expand the regulations to address additional plant operating modes, such as extended shutdown, refueling, or other potential situations. The range and implications of such event should be explored, perhaps quantifying their risk. The results of this exploration should then be reviewed relative to 10CFR50 to see if changes to the regulations are appropriate. Having such regulations would increase assurance that all significant contributors to risk have been identified and controlled.

Regulatory Position: General - Safety terminology

Potential Action: Clarify

Comments: Various terms which include the word "safety" in them have been used liberally in the regulations and subsidiary positions since the beginning of regulation with the Atomic Energy Act of 1954. The more common ones are: "safety-related," "important to safety," "nonsafety-related," "safety function," "safety margin," and several others. There was considerable confusion surrounding these terms for many years, particularly when issues arose which might involve backfitting or design decisions. The confusion was such that in 1984, then-Director of Nuclear Reactor Regulation, H. Denton issued Generic Letter 84-01 to all licensees and applicants redefining the terms. That guidance appears to hold today as the last documented statement in that regard. Yet there still appears to be confusion within the reactor safety community, including the NRC staff.

The potential action is somehow to clarify or reassert the definitions and appropriate uses of common terms containing the word "safety."

Regulatory Position: General - Currency of regulations

Potential Action: Expand

Comments: One perceived difficulty in ensuring effective, efficient regulation is the extent to which regulatory requirements have become highly intertwined and outdated. Despite efforts to keep regulation current and self-consistent, the body of regulatory positions has become too complex to manage. In many cases requirements have been imposed or negotiated through mechanisms (e.g., Generic Letters, Orders) other than formal rulemaking. In other cases the regulations have been changed in response to a specific incident (e.g., TMI, Brown's Ferry), focusing on existing reactors and imposing specific implementation schedules. In both cases, the intent is usually to harmonize all relevant regulatory positions at some later time, but in practice subsequent events combined with finite resources often delay self-consistency for years or indefinitely.

The potential action is to expand the regulations by adopting a "sunset" provision which forces periodic review of regulatory positions at some pre-specified time after their initial adoption. The review would address the continuing need for the position and any recommended changes. The outcomes of such a review might be a reaffirmation of the continuing need, a proposed revision, or a cessation of effectiveness. To a major extent this process is practiced informally during the normal course of regulation. The suggestion here is to formalize this process more, thus motivating more regular, higher level attention.

4. RECOMMENDATIONS FOR FURTHER ACTION

REFERENCES

1. "Policy and Planning Guidance 1987," NUREG-0885 Issue 6, US Nuclear Regulatory Commission, September 1987.

2. V. Stello to NRC Directors, "Program Guidance for 1987-88," November 3, 1986.

3. "Public Notice of Availability of Program Plan to Review Effectiveness of LWR Regulatory Requirements in Limiting Risk," Federal Register, Vol. 49, No. 193, October 3, 1984.

4. "Identification of Regulatory Requirements That Have Marginal Importance to Safety," W. Minners NRR to A. Tse RES, June 27, 1985.

5. M.F. Mullen et al, "Review of Light Water Reactor Regulatory Requirements," NUREG/CR-4330, Pacific Northwest Laboratory

   a. Volume 1 - Identification of Regulatory Requirements that May Have Marginal Importance to Risk, April 1986.

   b. Volume 2 - Assessment of Selected Regulatory Requirements that May Have Marginal Importance to Risk. Reactor Containment Leakage Rates, Main Steam Isolation Valve Leakage Control Systems, Fuel Design Safety Reviews. June 1986.

   c. Volume 3 - Assessment of Selected Regulatory Requirements that May Have Marginal Importance to Risk. Postaccident Sampling System, Turbine Missiles, Combustible Gas Control, Charcoal Filters. May 1987.

6. "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements," F. Miraglia NRR to all holders of Operating License or Construction Permit, Generic Letter 87-11, June 19, 1987.

7. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, June 1987.

APPENDIX A

AGENDA AND PREPARATORY MATERIALS
FOR A TYPICAL STRUCTURED INTERVIEW

EFFECTIVENESS OF LWR REQUIREMENTS
IN LIMITING RISK

AGENDA

1. Scope: Human Factors, Operator Licensing, Performance Evaluation and Quality Assurance. Concerns issues of human performance and organizational effectiveness. Concerns the qualifications, training and performance of plant operators, plant procedures, operator-machine interface, test programs and personnel policies. Concerns the development and implementation of programs to assure the quality and reliability of design, fabrication, construction, testing and operation.

2. Specific regulatory positions under consideration (using REGIS printouts)

3. Regulations/positions which are candidates for elimination, revision or replacement based on their impact on safety.

   3.1 Regulation/position
   3.2 Impact on safety
   3.3 Burden on NRC staff
   3.4 Specific action suggestion (eliminate, revise, replace, etc.)
   3.5 Reason

4. Implications for licensing future reactors

5. Additional views/comments on

   - unnecessary burdens on staff of operating reactors
   - relevant knowledge/suggestions from your experience outside your current branch
   - our approach for gathering information
   - anything else.

HUMAN FACTORS, OPERATOR LICENSING

Prior . gestions from NRR (June 1985)

1. Standard Review Plan Section 13.2 and reference to Item I.A.3.1 of NUREG-0737 related to simulator examinations for all licensees. Revise SRP to eliminate reference to I.A.3.1; replace with position from SECY-82-232 requiring simulator exams at sites with plant-specific simulators. Has this suggestion been further outmoded by requirements for all plants to have a plant-specific simulator?

2. Standard Review Plan Section 13.2 and reference to NUREG-0094 "NRC Operator Licensing Guide." Replace references to NUREG-0094 with updated NUREG-1021 "Operator Licensing Examiner Standards" (10/83).

3. Standard Review Plan Section 13.2 and reference to Item I.A.2.1 of NUREG-0737 related to control manipulations by licensed operators. A major update is needed to reflect changes in emergency procedures, ATWS, PTS and multiple failure scenarios.

Prior Suggestions in NUREG/CR-4330 (April 1986)

1. 10 CFR 50 Appendix R disallows credit for operator actions to mitigate the effects of plant fires. Modify Appendix R to allow credit for fire suppression and plant control actions for which specific emergency procedures and drills exist.

2. 10 CFR 50 Appendix R requires permanently installed emergency lighting. Modify Appendix R to allow credit for hand-held emergency lighting in some areas of the plant.

3. Generic Letter 82-28 related to reactor vessel level indication system. Eliminate requirements for the system. Emergency operating procedures and training are more effective than RVLIS, whose performance has generally been poor to date, costly and nonbeneficial to risk.

4. Standard Review Plan Section 13.2.1 related to simulator examinations. See above.

5. Regulatory Guide 1.78 regarding habitability of control room during hazardous chemical release. Delete requirement for sulfur and ammonia detectors based on absence of these hazards around plant sites.

6. Regulatory Guide 1.97 regarding instrumentation to monitor accidents. Eliminate or revise to enhance flexibility associated with post-accident monitoring.

7. Standard Review Plan Section 13.2 and related reference to NUREG-0094 on operator requalification examinations. See above.

EFFECTIVENESS OF LWR REQUIREMENTS
IN LIMITING RISK

1. Scope: Human Factors, Operator Licensing, Performance Evaluation and Quality Assurance. Concerns issues of human performance and organizational effectiveness. Concerns the qualifications, training and performance of plant operators, plant procedures, operator-machine interface, test programs and personnel policies. Concerns the development and implementation of programs to assure the quality and reliability of design, fabrication, construction, testing and operation.

2. Specific regulatory position utilized

☐ 10CFR Parts 19.12, 30.53, 50.34, 50.34a, 50.34b, 50.40b, 50.54, 50.54i, 50.54j, 50.54k, 50.54l, 50.54m, 50.55a, 50.55e, Part 50 App. B, Part 50 App. J, 55.21, 55.22, 55.23, Part 55 App. A

☐ General Design Criteria 1, 19

☐ Standard Review Plan Sections 13.1.1, 13.1.2-3, 13.2.1, 13.2.2, 13.4, 13.5.1, 13.5.2, 14.1, 14.2, 14.3, 17.1, 17.2, 18.0, 18.1, 18.2

☐ Regulatory Guides 1.8, 1.18, 1.20, 1.26, 1.28, 1.29, 1.30, 1.33, 1.37, 1.38, 1.39, 1.41, 1.52, 1.56, 1.58, 1.64, 1.68, 1.72, 1.74, 1.79, 1.80, 1.88, 1.94, 1.95, 1.108, 1.116, 1.123, 1.128, 1.139, 1.144, 1.146, 1.149

☐ NUREGS 0094, 0578, 0660, 0694, 0700, 0718, 0737, 0899

☐ Generic Letters ______

☐ Orders ______

☐ Other ______

3. Are there any regulations/positions which the NRC should consider eliminating, revising or replacing based on their impact on safety

☐ Yes; Proceed to 3.1 ☐ No; Proceed to 3.2 ☐ Uncertain

3.1 Complete the following (use additional forms if necessary)

Cite regulation/position

Estimate its impact on safety (check one)	Estimate its burden on NRC staff (check one)	Specific actions suggested
☐ Marginal positive	☐ High (>10 staff-years/year)	☐ Eliminate
☐ None	☐ Medium (1-10)	☐ Modify; How? ______
☐ Negative	☐ Low (<1)	☐ Replace; With what? ______
☐ Uncertain	☐ Uncertain	

3.2 Provide the reason for your view

4. With respect to the licensing of future reactors, of the regulations/positions listed in 2 above, which would you

☐ Carry over? ______

☐ Leave to licensees' discretion? ______

☐ Eliminate? ______

☐ Augment with additional requirements/guidance? ______

5. Provide any additional views/comments on back.

 Search

Human or [illegible] or Performance
or Quality

SRP Section	SRP Title	Primary Review Branch	Secondary Review Branch	References To 10 CFR	NUREG References	Regulatory Guide References And Titles	
13.01.01	Management and Technical Support Organization	Performance Evaluation	Radiation Protection	50.40b	NUREG-0694, NUREG-0718, NUREG-0737	RG 1.008	QUALIFICATION AND TRAINING OF PERSONNEL
13.01.02-03	Operating Organization	Performance Evaluation		50.40b, 50.54j, 50.54k, 50.54l, 50.54m	NUREG-0694, NUREG-0737	RG 1.008 RG 1.033	QUALIFICATION AND TRAINING OF PERSONNEL QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)
13.02.01	Reactor Operator Training	Human Factors Assessment	Operator Licensing	50.54i, 50.54j, 50.54k, 50.54l, 50.54m, 55.21, 55.22, 55.23, Part 55 App. A	NUREG-0094, NUREG-0737, NUREG-0718	RG 1.008 RG 1.149	QUALIFICATION AND TRAINING OF PERSONNEL SIMULATORS FOR OPERATOR TRAINING
13.02.02	Training for Non-Licensed Plant Staff	Human Factors Assessment		19.12, 50.34a, 50.34b, 50.40b	NUREG-0660, NUREG-0737	RG 1.008 RG 1.149	QUALIFICATION AND TRAINING OF PERSONNEL SIMULATORS FOR OPERATOR TRAINING
13.04	Operational Review	Performance Evaluation		50.40b	NUREG-0737	RG 1.008 RG 1.033	QUALIFICATION AND TRAINING OF PERSONNEL QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)
13.05.01	Administration Procedures	Performance Evaluation		50.40b, 50.54	NUREG-0578, NUREG-0694, NUREG-0737	RG 1.033	QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)
13.05.02	Operating and Maintenance Procedures	Human Factors Assessment		50.34a, 50.34b	NUREG-0737, NUREG-0899	RG 1.033	QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)
14.01	Initial Plant Test Programs	Quality Assurance Branch					
14.02	Initial Plant Test Program - Final Safety Analysis Report	Quality Assurance Branch		30.53, 50.34b, Part 50 App. B, Part 50 App J	NUREG-0660, NUREG-0694, NUREG-0737	RG 1.018 RG 1.020 RG 1.030 RG 1.037 RG 1.041	**WITHDRAWN** VIBRATION ASSESSMENT PROGRAM FOR REACTOR INTERNALS INSTRUMENTATION & ELECTRIC EQUIPMENT QUALITY ASSURANCE FLUID SYSTEM CLEANING QUALITY ASSURANCE ELECTRIC POWER SYSTEM PRE-OPERATIONAL TEST

SRP Section	SRP Title	Primary Review Branch	Secondary Review Branch	References To 10 CFR	NUREG References	Regulatory Guide References And Titles
						RG 1.052 ENGINEERED SAFETY FACTORS AIR FILTER SYSTEM DESIGN CRITERIA RG 1.056 BWR WATER PURITY RG 1.068 INITIAL TESTING PROGRAMS RG 1.072 FIBERGLASS REINFORCED SPRAY POND PIPING RG 1.079 PRE-OPERATIONAL EMERGENCY CORE COOLING SYSTEM TESTING RG 1.080 **WITHDRAWN** RG 1.095 CONTROL ROOM CHLORINE PROTECTION RG 1.108 EMERGENCY DIESEL GENERATOR PERIODIC TESTING-ELECTRIC SYSTEMS RG 1.116 QUALITY ASSURANCE FOR MECHANICAL EQUIPMENT & SYSTEMS RG 1.128 STATION STORAGE BATTERY INSTALLATION & DESIGN RG 1.139 RESIDUAL HEAT REMOVAL GUIDANCE
14.03	Standard Plant Designs, Initial Test Program - Final Design Approval (FDA)	Quality Assurance Branch				
17.01	Quality Assurance during the Design and Construction Phases	Quality Assurance Branch	Engineering and Systems Technology	Part 50 App B, 50.55a, 50.55e, 50.34, GDC 1		RG 1.008 QUALIFICATION AND TRAINING OF PERSONNEL RG 1.026 QUALITY CLASSES FOR WATER, STEAM & RADIATION COMPONENTS RG 1.028 QUALITY ASSURANCE PROGRAM REQUIREMENTS RG 1.029 SEISMIC DESIGN CLASSIFICATION RG 1.030 INSTRUMENTATION & ELECTRIC EQUIPMENT QUALITY ASSURANCE RG 1.037 FLUID SYSTEM CLEANING QUALITY ASSURANCE RG 1.038 QUALITY ASSURANCE FOR WAREHOUSING RG 1.039 HOUSEKEEPING REQUIRMENTS RG 1.058 INSPECTION & TESTING PERSONNEL QUALIFICATION RG 1.064 PLANT DESIGN QUALITY ASSURANCE RG 1.074 QUALITY ASSURANCE TERMS & DEFINITIONS RG 1.088 MAINTENENCE OF QUALITY ASSURANCE RECORDS RG 1.094 QUALITY ASSURANCE FOR STEEL & CONCRETE CONSTRUCTION RG 1.116 QUALITY ASSURANCE FOR MECHANICAL EQUIPMENT & SYSTEMS RG 1.123 PROCURMENT QUALITY ASSURANCE RG 1.144 AUDITING OF QUALITY ASSURANCE PROGRAMS RG 1.146 QUALITY ASSURANCE AUDITOR QUALIFICATION
17.02	Quality Assurance during the Operations Phase	Quality Assurance Branch	Engineering and Systems Technology	Part 50 App B, 50.34, 50.55a, GDC 1		RG 1.008 QUALIFICATION AND TRAINING OF PERSONNEL RG 1.026 QUALITY CLASSES FOR WATER, STEAM & RADIATION COMPONENTS RG 1.028 QUALITY ASSURANCE PROGRAM REQUIREMENTS RG 1.029 SEISMIC DESIGN CLASSIFICATION RG 1.030 INSTRUMENTATION & ELECTRIC EQUIPMENT QUALITY ASSURANCE RG 1.037 FLUID SYSTEM CLEANING QUALITY ASSURANCE RG 1.038 QUALITY ASSURANCE FOR WAREHOUSING RG 1.039 HOUSEKEEPING REQUIRMENTS RG 1.058 INSPECTION & TESTING PERSONNEL QUALIFICATION RG 1.064 PLANT DESIGN QUALITY ASSURANCE RG 1.074 QUALITY ASSURANCE TERMS & DEFINITIONS RG 1.088 MAINTENENCE OF QUALITY ASSURANCE RECORDS RG 1.094 QUALITY ASSURANCE FOR STEEL & CONCRETE CONSTRUCTION

SRP Section	SRP Title	Primary Review Branch	Secondary Review Branch	References To 10 CFR	NUREG References	Regulatory Guide References And Titles
						RG 1.116 QUALITY ASSURANCE FOR MECHANICAL EQUIPMENT & SYSTEMS RG 1.123 PROCURMENT QUALITY ASSURANCE RG 1.144 AUDITING OF QUALITY ASSURANCE PROGRAMS RG 1.146 QUALITY ASSURANCE AUDITOR QUALIFICATION
18.00	Human Factors Engineering/Standard Review Plan Development	Human Factors Assessment				
18.01	Control Room	Human Factors Assessment		GDC 19	NUREG-0700	
18.02	Safety Parameter Display Sy^tem (SPDS)	Human Factors Assessment	Plant Systems Instrumentation and Control Systems	GDC 19	NUREG-0660, NUREG-0737, NUREG-0700	

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