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REMARKS

This previous Central File material can now be made publicly available.

MATERIAL RELATED TO ACRS MEETING NO. 196

CC (LIST ONLY) JEAN RATAJE, PDR L STREET

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MATERIAL RELATED TO CRGR MEETING NO. 196
TO BE MADE PUBLICLY AVAILABLE

1. MEMO FOR J. TAYLOR FROM E. JORDAN DATED 1/14/91
SUBJECT: MINUTES OF CRGR MEETING NUMBER 196
INCLUDING THE FOLLOWING ENCLOSURES WHICH WERE NOT PREVIOUSLY RELEASED:

a. ENCLOSURE 2
A SUMMARY OF DISCUSSIONS OF A ~~PROPOSED~~ Briefing on Improved Standard Tech. Specs and Four Request for Waiver of CRGR Review regarding Specific Line Item TSI's

b. ENCLOSURE ________
A SUMMARY OF DISCUSSIONS OF A PROPOSED Technical Position on Waste Form

c. ENCLOSURE ________
A SUMMARY OF DISCUSSIONS OF A PROPOSED

2. MEMO FOR E. JORDAN FROM F. Miraglia DATED undated
FORWARDING REVIEW MATERIALS ON A ~~PROPOSED~~ New Standard Technical Specifications (STS)

3. MEMO FOR E JORDAN FROM F. Miraglia DATED 8/14/90
FORWARDING REVIEW MATERIALS ON A PROPOSED GL on the Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from TS's

4. MEMO FOR E JORDAN FROM ________________ DATED ________
FORWARDING REVIEW MATERIALS ON A PROPOSED

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

January 14, 1991

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING NO. 196

The Committee to Review Generic Requirements (CRGR) met on Wednesday, December 12, 1990 from 1:00-5:00 p.m. A list of attendees at the meeting is enclosed (Enclosure 1). The following items were discussed at the meeting:

1. E. Rossi, J. Calvo, M. Reinhart and T. Dunning of NRR provided a briefing on improved standard technical specifications and four requests for waiver of CRGR review regarding specific line item technical specification improvements.

   With regard to the improved standard technical specifications, which would be reviewed at a future meeting, the CRGR provided a number of questions and comments for staff consideration.

   With regard to the waiver requests, the disposition was as follows:

   (a) Proposal to remove testing requirements for BWR scram accumulator check valves.

   This proposal was withdrawn by the staff.

   (b) Proposal to remove lists of acceptable response times with regard to response time testing.

   The CRGR requested a full review of this matter and the staff agreed to prepare a review package.

   (c) Proposal to remove the schedule for removal of reactor vessel surveillance specimens.

   The CRGR agreed that there was no need for further formal review of this item.

   (d) Proposal to remove lists of components to which certain requirements apply.

   The CRGR agreed that there was no need for further formal review of this item.

   This matter is discussed in Enclosure 2.

2. J. Greeves, J. Surmeier and M. Tokar of NMSS provided a briefing on a proposed technical position on waste form. The CRGR agreed with the NMSS judgment that formal CRGR review of this item was not needed. This matter is discussed in Enclosure 3.

~~9103010103~~ 2pp

In accordance with the EDO's July 18, 1983 directive concerning "Feedback and Closure of CRGR Reviews," a written response is required from the cognizant office to report agreement or disagreement with CRGR recommendations in these minutes. The response, which is required within five working days after receipt of these minutes, is to be forwarded to the CRGR Chairman and if there is disagreement with CRGR recommendations, to the EDO for decisionmaking.

Questions concerning these meeting minutes should be referred to Dennis Allison (492-4148).

Original Signed by:
E. L. Jordan

Edward L. Jordan, Chairman
Committee to Review Generic
Requirements

Enclosures:
As stated

cc: Commission (5)
SECY
J. Lieberman
P. Norry
D. Williams
Regional Administrators
CRGR Members

Distribution:
Central File (w/o encl.)
PDR/DCS (NRC/CRGR) (w/o encl.)

P. Kadambi	CRGR C/F
CRGR S/F	M. Taylor
J. Sniezek	E. Rossi
J. Calvo	E. Sullivan
G. Thomas	R. Bangert
J. Surmeier	D. Ross
E. Jordan	D. Allison
J. Conran	

DPA

OFC	CRGR/AEOD	DD:AEOD	C/CRGR/AEOD			
NAME	DAllison:slm	DRoss	EJordan			
DATE	1/11/91	1/ /91	1/14/91			

Enclosure 1

ATTENDANCE LIST

CRGR Meeting No. 196

December 12, 1990

CRGR Members

E. Jordan
G. Arlotto
J. Moore
F. Miraglia
B. Sheron
L. Reyes

CRGR Staff

J. Conran
D. Allison

NRC Staff

E. Rossi
M. Reinhart
J. Calvo
T. Dunning
R. Lobel
J. Tsao
R. Emch
J. Surmeier
M. Reinhart
J. Greeves
N. Gill
M. Tokar
C. Harbuck

Enclosure 2 to the Minutes of CRGR Meeting No. 196
Briefing on Improved Standard Technical Specifications
and Four Request for Waiver of CRGR Review Regarding
Specific Line Item Technical Specification Improvement

December 12, 1990

TOPIC/CONCLUSIONS

E. Rossi, J. Calvo, M. Reinhart and T. Dunning of NRR provided a briefing on improved standard technical specifications and four requests for waiver of CRGR review regarding specific line item technical specification improvements.

(1) The improved standard technical specifications were to be issued for comment in the near future. The package would be provided to the CRGR for information at that time. It would consist of about 15,000 pages, including about 4,000 technical specification changes. After subsequent consideration of comments and appropriate revision, the package would be sent to CRGR for review.

It was noted that licensees' adoption of the new standard technical specifications would be voluntary. To the extent licensees did volunteer to adopt the new standards, NRC acceptance would be contingent upon adoption of an upgraded 10 CFR 50.59 review process as described in an industry document, NSAC-125. A one year trial program using this guidance was nearing completion.

It was noted that the CRGR would be interested in a briefing on the NSAC-125 program.

With regard to risk during shutdown modes, it was noted that, for the forthcoming improved standard technical specifications, the staff would have a basis for its decisions as to the modes for which each requirement would apply. However, the search for any new specifications that might be needed to reduce risk in shutdown modes would be completed later.

The specific line item improvements discussed below were related to the improved STS in that they would be included in the improved STS. However, they were really separate actions being taken now and in that sense they would be independent of the improved STS.

(2) Requests for waiver of CRGR review regarding specific line item technical specification improvements:

(a) Proposal to remove testing requirements for BWR scram accumulator check valves.

The CRGR had some comments and questions about this proposal. However, prior to the meeting the staff had decided to withdraw the request.

(b) Proposal to remove lists of acceptable response times with regard to response time testing.

The CRGR had a number of comments and questions on this proposal and requested a full CRGR review. Such review could be deferred until CRGR review of the improved STS, at the staff's discretion. The staff agreed to provide a CRGR review package and indicated that it did not intend to wait until review of the STS.

The CRGR requested that the staff address the question of how it makes the finding that there will be no decrease in safety as a result of removing the requirements from the TS and placing them in other documents under the control of the 10 CFR 50.59 in view of weaknesses that have been noted in that review process.

(c) Proposal to remove the reactor vessel surveillance specimen removal schedule.

The CRGR noted that this item is also covered by rule, under Appendix H to 10 CFR 50. The CRGR agreed that there was no need for further formal review of this matter.

(d) Proposal to remove lists of components to which certain requirements apply.

The CRGR agreed that there was no need for further formal review of this item.

A copy of the handout materials used by the staff in its presentation is provided as an attachment to this enclosure.

BACKGROUND

1. A package of background material related to the improved standard technical specifications was transmitted by a memorandum for E. Jordan from F. Miraglia (undated) sent on December 7, 1990. The enclosures included:

-- Interim policy statement on technical specification improvements, 2/6/87.

-- Letters to owners groups on relocation of requirements, 5/9/88.

-- SECY-88-304 on reducing testing at power, 10/26/88.

-- SECY-90-366 on status of technical specification improvement, 10/29/90.

2. Waiver requests were transmitted as follows:

a. Memorandum for E. Jordan from F. Miraglia, dated August 23, 1990 regarding removal of testing requirements for BWR scram accumulator check valves from technical specifications.

b. Memorandum for E. Jordan from F. Miraglia, dated August 23, 1990 regarding removal of response time limits from technical specifications.

c. Memorandum for E. Jordan form F. Miraglia, dated August 14, 1990 regarding removal of schedule for removal of reactor vessel material specimens from technical specifications.

d. Memorandum for E. Jordan from F. Miraglia, dated November 16, 1990 regarding removal of component lists from technical specifications.

COMMITTEE FOR REVIEW OF GENERIC REQUIREMENTS

NEW STANDARD TECHNICAL SPECIFICATIONS (STS)

MARK REINHART

WEDNESDAY, DECEMBER 12, 1990

ATTACHMENT TO
ENCLOSURE 2

INFORMATION BRIEFING ON NEW STANDARD TECHNICAL SPECIFICATIONS (STS)

• OVERVIEW OF PROGRAM AND PROGRESS	TODAY
• RELEASE FINAL DRAFT FOR YOUR INFORMATION	JAN 91

CHRONOLOGY: STANDARD TECHNICAL SPECIFICATIONS (STS)

- Background

Commission's Interim Policy Statement	Feb 87
"Split Report"	May 88
Owners Groups Proposed new STS	Mar 89 to Jun 89
Staff's Review and Discussions with Owners Groups	Apr 89 to Dec 90

- Progress

Staff to issue Final Draft new STS and their Bases	Jan 91
Owners Groups' and NRC staff's final review	

- Future

Apply lessons learned from lead plant conversions to new STS	
Issue new STS and their Bases	Spring 91

EXTENT OF PARTICIPATION IN PROGRAM

- INDUSTRY PARTICIPATION (30 PERSONS)
  - NUMARC
  - NSSS OWNERS GROUPS
  - LEAD PLANT LICENSEES
  - OTHER LICENSEES
- NRC STAFF PARTICIPATION (65 PERSONS)
  - TECHNICAL SPECIFICATIONS BRANCH
  - NRR TECHNICAL BRANCHES (INCLUDING RISK AND HUMAN FACTORS)
  - PROJECTS
  - REGIONS
  - TECHNICAL TRAINING CENTER
- NRC CONTRACTORS (25 PERSONS)
  - LAWRENCE LIVERMORE NATIONAL LABORATORY
  - IDAHO NATIONAL ENGINEERING LABORATORY
  - PACIFIC NORTHWEST LABORATORIES
  - SCIENCE APPLICATIONS INTERNATIONAL CORPORATION

LEAD PLANT CONVERSIONS TO NEW STS

North Anna 1 and 2	Westinghouse
Crystal River 3	Babcock and Wilcox
San Onofre 2 and 3	Combustion Engineering
Hatch 2	GE BWR-4
Grand Gulf 1	GE BWR-6

CONTENTS OF NEW STS

1.0 USE AND APPLICATION

- 1.1 DEFINITIONS
- 1.2 LOGICAL CONNECTORS
- 1.3 COMPLETION TIMES
- 1.4 FREQUENCY
- 1.5 OPERABILITY

2.0 SAFETY LIMITS

LIMITING CONDITIONS FOR OPERATION
AND SURVEILLANCE REQUIREMENTS

3.0 APPLICABILITY
3.1 REACTIVITY CONTROL SYSTEMS
3.2 POWER DISTRIBUTION LIMITS
3.3 INSTRUMENTATION
3.4 REACTOR COOLANT SYSTEM
3.5 EMERGENCY CORE COOLING SYSTEMS
3.6 CONTAINMENT
3.7 PLANT SYSTEMS
3.8 ELECTRICAL
3.9 REFUELING
3.10 SPECIAL OPERATIONS (BWR'S)

4.0 DESIGN FEATURES

5.0 ADMINISTRATIVE CONTROLS

HIGHLIGHTS OF CHANGES

- Technical Changes

  Relocated 40% of requirements to licensee controlled documents
    Licensees to provide controls for relocated requirements
  Reduced Surveillance Testing
  Line Item Improvements

- Risk Insights
  Split (3 Criteria + risk insights)
  Topical Reports on Instrumentation Completion Times and
    Surveillance Frequencies
  SAIC evaluation

- Human Factors
  Writers Guide

SUMMARY OF IMPROVEMENTS

- Focused on operational safety
- More Operator oriented
- Streamlined LCO's and SR's
- High degree of consistency within each and among all STS
- Bases provide
  - Reasons for LCO and SR requirements
  - Link with Safety Analysis
- Promote better understanding of technical specifications
- Allow more efficient use of NRC and industry resources

Enclosure 3 to the Minutes of CRGR Meeting No. 196

Briefing on Proposed Technical Position

on Waste Form

December 12, 1990

TOPIC/CONCLUSION

J. Greeves, J. Surmeier and M. Tokar of NMSS provided a briefing on a proposed technical position on waste form.

The purposes of the briefing were to inform the CRGR of a significant action in accordance with a previous CRGR request and to confirm the NMSS judgment that a full CRGR review would not be needed.

The proposed action would issue new criteria for concrete used to encapsulate low level waste. The new criteria would address problems and weaknesses found using current practice. (Other waste forms such as canisters and organic materials had previously been addressed.)

The CRGR agreed that CRGR review was not needed for this item.

BACKGROUND

The draft technical position was described in a memorandum for E. Jordan from R. Bernero, dated December 6, 1990. The enclosures included:

1. Draft technical position.

2. Letter from Moeller, ACNW, to Carr, NRC, dated 9/6/90.

3. Memorandum for Bangert, NMSS, from Treby, OGC, dated 6/18/90.

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

MEMORANDUM FOR: Edward L. Jordan, Director
Office for Analysis and Evaluation of
Operational Data

FROM: Frank J. Miraglia, Jr., Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: CRGR BRIEFING ON THE NEW STANDARD TECHNICAL SPECIFICATIONS (STS)

NRR is scheduled to brief CRGR on the new Standard Technical Specifications on December 12, 1990. It is anticipated that a final draft of the new STS will be issued to the owners groups for comment in the very near future. It is not necessary to have reviewed the new STS prior to the briefing since this briefing is intended only to introduce the new STS to CRGR. It is anticipated that future meetings will be scheduled at which the major issues can be discussed in detail, if desired.

In order to provide some background information for the first briefing, we are providing the following documents to CRGR members and staff:

1. Commission (interim) Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.
2. Letters to the owners group chairmen providing lists of requirements which may be relocated from the STS, May 9, 1988.
3. SECY-88-304 Staff Actions to Reduce Testing at Power, October 26, 1988.
4. SECY-90-366 Report on the Status of the Technical Specifications Improvement Program, October 29, 1990.

The contact for this effort is Mr. Richard Lobel (x21185). This effort is sponsored by Charles E. Rossi, Director, Division of Operational Events Assessment.

We look forward to introducing CRGR to the large amount of work which has been done by the staff and the industry to improve the technical specifications.

Frank J. Miraglia

Frank J. Miraglia, Jr., Deputy Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated

~~9012130139~~ 34pp

52 FR 3788 (February 6, 1987)

[7590-01]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Commission Policy Statement on
Technical Specification Improvements
for Nuclear Power Reactors

AGENCY: Nuclear Regulatory Commission.

ACTION: Interim Policy Statement.

SUMMARY: This statement presents the policy of the Nuclear Regulatory Commission (NRC) with respect to the scope and purpose of Technical Specifications for nuclear power plants as required by 10 CFR 50.36. It establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. It encourages licensees to implement a voluntary program to update their Technical Specifications to be consistent with revised vendor-specific Standard Technical Specifications (STS) to be developed by the industry based on these criteria and subject to NRC Staff approval. The Policy Statement also identifies mechanisms to be used by the NRC and industry to control changes to those items removed from Technical Specifications. The Policy Statement is expected to produce an improvement in the safety of nuclear power plants through the development of more operator-oriented Technical Specifications, improved Technical Specification Bases, reduced action statement-induced plant transients, and more efficient use of NRC and industry resources.

DATE: This Interim Policy Statement is effective upon issuance. However, the public is invited to submi. comments by March 23, 1987. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before this date. On the basis of the submitted comments, the Commission will determine whether to modify the Policy Statement before issuing it as final.

FOR FURTHER INFORMATION CONTACT: David C. Fischer, Technical Specifications Coordination Branch, Division of Human Factors Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, telephone (301) 492-7924.

SUPPLEMENTARY INFORMATION:

I. BACKGROUND

Section 182a. of the Atomic Energy Act of 1954, as amended (42 U.S.C. 2232), mandates the inclusion of Technical Specifications in licenses for the operation of production and utilization facilities. The Act requires that Technical Specifications include information of the amount, kind, and source of special nuclear material, the place of use, and the specific characteristics of the facility. That section also indicates that Technical Specifications should contain such information as the Commission may by rule deem necessary to enable it to find that the utilization of special nuclear material will be in accord with the common defense and will provide adequate protection of public health and safety. Finally, that section requires Technical Specifications to be made a part of any license issued.

Section 50.36, "Technical Specifications," which implements Section 182a. of the Atomic Energy Act, was promulgated by the Commission on December 17, 1968 (33 FR 18610). This rule delineates requirements for determining the contents of Technical Specifications. Technical Specifications set forth the specific characteristics of the facility and the conditions for its operation that are required to provide adequate protection to the health and safety of the public. Specifically, 10 CFR 50.36 requires that:

"Each license authorizing operation of a production or utilization facility of a type described in §50.21 or §50.22 will include Technical Specifications. The Technical Specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to §50.34. The Commission may include such additional Technical Specifications as the Commission finds appropriate."

Technical Specifications cannot be changed by licensees without prior NRC approval. However, since 1969, there has been a trend towards including in Technical Specifications not only those requirements derived from the analyses and evaluation included in the safety analysis report but also essentially all other Commission requirements governing the operation of nuclear power reactors. This extensive use of Technical Specifications is due in part to a lack of well defined criteria (in either the body of the rule or in some other regulatory document) for what should be included in Technical Specifications. This has contributed to the volume of Technical Specifications and to the several fold increase, since 1969, in the number of license amendment applications to effect changes to the Technical Specifications. It has diverted both staff and licensee attention from the more important requirements in these documents to the extent that it has resulted in an adverse but unquantifiable impact on safety.

On March 30, 1982, the NRC published in the Federal Register (47 FR 13369) a proposed amendment to its regulations, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The proposed amendment would have revised §50.36, "Technical Specifications," to establish a new system of specifications divided into two general categories. Only those specifications contained in the first general category as Technical Specifications would have become part of the operating license and require prior NRC approval for any changes. Those specifications contained in the second general category would have become supplemental specifications and would not require prior NRC approval for most changes. The NRC review of the first general category of specifications would have been the same as currently performed for Technical Specifications changes, which are amendments to the operating license. For the second category, supplemental specifications, the licensee would have been allowed to make changes within specified conditions without prior NRC approval. The NRC would have reviewed these changes when they were made and would have done so in a manner similar to that currently used for reviewing design changes, tests, and experiments performed under the provisions of 10 CFR 50.59.

Because of difficulties with defining the criteria for dividing the Technical Specifications into the two categories of the proposed rule and other higher priority licensing work, the rule change was deferred.

In the past several years the nuclear industry and the NRC Staff have been studying the question of whether improvement to the current system of establishing Technical Specification requirements for nuclear power plants is needed. The two most recent studies of this issue were performed by an NRC task group known as the Technical Specifications Improvement Project (TSIP) and a Subcommittee of the Atomic Industrial Forum's (AIF) Committee on Reactor Licensing and Safety.[1] The overall conclusion of these studies was that many improvements in the scope and content of Technical Specifications are needed, and that a joint NRC and Industry program should be initiated to implement these improvements. Both of these groups made specific recommendations which are summarized as follows:

1) The NRC should adopt the criteria for defining the scope of Technical Specifications proposed in the AIF and TSIP reports. Those criteria should then be used by the NRC and each of the nuclear steam supply

[1] SECY-86-10, "Recommendations for Improving Technical Specification," dated January 13, 1986, contains both "Recommendations for Improving Technical Specifications," NRC Technical Specifications Improvement Project, September 30, 1985, and "Technical Specifications Improvements," AIF Subcommittee on Technical Specifications Improvements, October 1, 1985.

system vendor owners groups to completely rewrite and streamline the existing Standard Technical Specifications (STS). This process would result in many requirements being transferred from control by Technical Specification requirements to control by other mechanisms [e.g., the Final Safety Analysis Report (FSAR), Operating Procedures, Quality Assurance (QA) Plan] which would not require a license amendment or prior NRC approval when changes are needed. The new STS should include greater emphasis on human factors principles in order to add clarity and understanding to the text of the STS. The new STS should also provide improvements to the Bases Section of Technical Specifications which provides the purpose for each requirement in the specification.

2) A parallel program of short-term improvements in both the scope and substance of the existing Technical Specifications should be initiated in addition to developing a new STS as identified in (1) above.

II. DISCUSSION

The Commission recognizes the advantages of improved Technical Specifications. Clarification of the scope and purpose of Technical Specifications will provide useful guidance to both the NRC and industry and should serve as an important incentive for industry participation in a voluntary program to improve Technical Specifications. It will result in Technical Specifications that focus licensee's and the plant operator's attention on those plant conditions most important to safety and should also result in more efficient use of agency and industry resources.

The Policy Statement identifies three objective criteria for defining the scope of Technical Specifications. These criteria are intended to be consistent with the scope of Technical Specifications as stated in the Statement of Consideration accompanying the current rule.

The Statement of Consideration discusses the scope of Technical Specifications as including the following:

"In the revised system, emphasis is placed on two general classes of technical matters: (1) those related to prevention of accidents, and (2) those related to mitigation of the consequences of accidents. By systematic analysis and evaluation of a particular facility, each applicant is required to identify at the construction permit stage, those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity. Such items are expected to be the subjects of Technical Specifications in the operating license."

33 FR 18610 (December 17, 1968). The first of these two general classes of technical matters to be included in Technical Specifications is captured by criterion (1) and to some extent criterion (2) in that they address systems and process variables that alert the operator to a situation when accident initiation is more likely. The second general class of technical matters is explicitly addressed and captured by criteria (2) and (3). By applying the three criteria contained in the Policy Statement a licensee should capture all of those specific characteristics of its facility and the conditions for its operation that are required to meet the principal operative standard in Section 182a. of the Atomic Energy Act, that is, that adequate protection is provided to the health and safety of the public.

The Commission recognizes that the three criteria carry with them a common theme of focusing on those requirements related to technical matters dealing with those features of a facility that are of controlling importance to safety. Since many of the requirements are of immediate concern to the health and safety of the public, the Policy Statement adopts, for the purpose of relocating requirements from Technical Specifications to other licensee-controlled documents, the subjective statement of the purpose of Technical Specifications expressed by an Atomic Safety and Licensing Appeal Board Portland General Electric Company (Trojan Nuclear Plant), ALAB-531, 9 NRC 263 (1979). There the Appeal Board interpreted Technical Specifications as being reserved for those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. The Commission wishes to emphasize that this Policy Statement is intended to be consistent with the language of Section 182a. of the Atomic Energy Act, 10 CFR 50.36, and previous interpretations of the regulations. It merely clarifies the scope and purpose of Technical Specifications by identifying criteria which can be used to establish, more clearly, the framework for Technical Specifications (i.e., identify those requirements derived from the analyses and evaluation included in the safety analysis report and which are of immediate concern to the health and safety of the public). It identifies requirements which should be retained in Technical Specifications and also describes a mechanism whereby other "additional" requirements can be identified and controlled through mechanisms other than Technical Specifications.

The Commission invites public comment on this Policy Statement and particularly invites comment on the statement of the purpose of Technical Specifications which introduces the text of the Policy Statement and on whether it would be beneficial for licensees to be able to modify related portions of their LCOs (such as containment systems) without having to apply the terms and provisions of the Policy Statement to all LCOs.

III. THE COMMISSION'S POLICY

The purpose of Technical Specifications is to impose those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety by establishing those conditions of operation which cannot be changed without prior Commission approval and by identifying those features which are of controlling importance to safety.

Licensees are encouraged to implement a program to upgrade their Technical Specifications consistent with this purpose. The Commission will entertain requests based on the criteria below (as clarified by the supporting discussion) for individual license amendments that evaluate all of the Limiting Conditions for Operation (LCOs) for an individual plant to determine which LCOs should be included in the Technical Specifications. The Commission does not intend that these criteria be used as the basis for relocation of individual LCOs. LCOs which fail to meet any one or more of the criteria below may be removed from the Technical Specifications and relocated to other licensee-controlled documents, such as the FSAR or licensee procedures. The criteria may be applied to either Standard or custom Technical Specifications. However, it is expected that each of the nuclear steam supply system vendor owners groups will undertake the development of revised STS based on this Policy Statement, and we encourage licensees to use the revised STS as the basis for their individual plant Technical Specifications. The NRC will give first priority in its Technical Specifications improvements efforts to the review and approval of the revised STS and the plant specific license amendment applications based on them. Approved short term Technical Specifications improvements will be included in the revised STS. The revised STS and individual license amendment requests that are submitted based on this Policy Statement should incorporate all terms and provisions of the Policy Statement.

The following criteria delineate those constraints on design and operation of nuclear power plants that are derived from the plant safety analysis report and belong in Technical Specifications in accord with 10 CFR 50.36 and the purpose of Technical Specifications stated above.

<u>Criterion 1</u>: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary:

> <u>Discussion of Criterion 1</u>: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident.

This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage.

Criterion 2: A process variable that is an initial condition of a Design Basis Accident (DBA) or Transient Analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 2: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing Design Basis Accident and Transient Analyses. These analyses consist of postulated events, analyzed in the Final Safety Analysis Report (FSAR), for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the FSAR (or equivalent chapters) and are identified as Condition II, III, or IV events (ANSI N 18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

As used in Criterion 2, process variables are only those parameters for which specific values or ranges of values have been chosen as reference bounds in the Design Basis Accident or Transient Analyses and which are monitored and controlled during power operation such that process values remain within the analysis bounds.

The purpose of this criterion is to capture those process variables that have initial values assumed in the Design Basis Accident and Transient Analyses, and which are monitored and controlled during power operation. So long as these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated Design Basis Accident or Transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequence of the Design Basis Accident or Transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's Design Basis Accident and Transient Analyses, as presented in Chapters 6 and 15 of the plant's Final Safety Analysis Report (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to Design Basis Accidents and Transients limits the consequences of these events to within the appropriate acceptance criteria.

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function.

In addition to those structures, systems, and components captured by the above criteria, it is the Commission's policy that licensees retain in their Technical Specifications LCOs, action statements, and Surveillance Requirements for the following systems (as applicable) which operating experience and probabilistic risk assessment have generally shown to be important to public health and safety:

- Reactor Core Isolation Cooling (RCIC)/Isolation Condenser,
- Residual Heat Removal (RHR),
- Standby Liquid Control (SBLC), and
- Recirculation Pump Trip (RPT).

The Commission recognizes that features of plant design and operation not addressed in the safety analysis report's Design Basis Accidents or Transient Analyses can, in some cases, be significant contributors to the plant's overall core melt probability and risk. As stated in 10 CFR 50.36, the Commission may include such additional Technical Specifications as the Commission finds appropriate. Based on this, and consistent with the Commission's Safety Goal and Severe Accident Policy Statements, the Commission finds that risk evaluations are an appropriate tool for defining requirements that should be retained in Technical Specifications where including such requirements is consistent with the purpose of Technical Specifications as defined above.

The Commission expects that owners groups, in preparing their proposals to streamline the Standard Technical Specifications, will utilize the available literature on risk insights and Probabilistic Risk Assessments (PRAs). This material should be employed to strengthen the technical bases for those requirements that remain in Technical Specifications, when applicable, and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk. Similarly, the Staff will also employ risk insights and PRAs in evaluating the revised STS.

In some cases, plant-specific PRAs or risk surveys conducted, for example, pursuant to the Commission's Severe Accident Policy, may be available to licensees as they prepare license amendments to adopt the revised STS to their plant, or to streamline custom Technical Specifications under this Policy Statement. Where such PRAs or surveys are available, they should be used to strengthen the Bases and screen those Technical Specifications to be relocated, as suggested above. Where such plant-specific risk surveys are unavailable, licensees should utilize the available literature on risk insights and PRAs, as described above. However, licensees need not await the performance of plant-specific PRA studies before availing themselves of this policy. As in the case of the revised STS discussed above, the Staff will also utilize risk insights and PRAs in evaluating the plant-specific submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue research in methods to make better use of risk and reliability considerations for defining future generic Technical Specification requirements.

Requirement(s) which would be relocated from Technical Specifications to another licensee-controlled document (e.g., the FSAR and 10 CFR 50.59, Operating Procedures, the QA Plan, or Fire Protection Plan) may be changed or deleted in conjunction with the filing of the revised STS or of individual license amendment request to implement this Policy Statement. The package containing the revised STS or the amendment request must contain a clear statement of the basis of the requirement(s) to be changed or deleted, a safety evaluation, and a statement that the change(s) has been reviewed by a multidisciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

When licensees submit amendment requests based on this Policy Statement, they should identify the location of, and controls for, the technical and administrative requirements of the removed Technical Specifications. The Staff will carefully review these submittals to ensure the accountability of each relocated requirement.

Appropriate surveillance requirements and action statements should be retained for each LCO which remains in the Technical Specifications. Each LCO, Action Statement, and Surveillance Requirement should have supporting Bases. The Bases should at a minimum address the following questions and cite references to appropriate licensing documentation (e.g., FSAR, Topical Report) to support the Bases.

1. What is the justification for the Technical Specification, i.e., which criterion requires it to be in the Technical Specifications?

2. What are the Bases for each Limiting Condition for Operation (LCO), i.e., why was it determined to be the lowest functional capability or performance level for the system/component in question necessary for safe operation of the facility and what are the reasons for the Applicable Operational Modes(s) for the LCO?

3. What are the Bases for each Action Statement, i.e., why should this remedial action be taken if the associated LCO cannot be met, how does this action relate to other Action Statements associated with the LCO, and what justifies continued operation of the system/component at the reduced state from the state specified in the LCO for the allowed time period?

4. What are the Bases for each Limiting Safety System Setting?

5. What are the Bases for each Surveillance Requirement and the surveillance interval specified, i.e., what specific functional requirement is the surveillance designed to verify, and why is this surveillance necessary at the specified frequency to assure that the system/component function is maintained, that facility operation will be within the safety limits, and that the LCO will be met?

NOTE: In answering these questions the Bases for each number (e.g., Trip Set point, Response Time, Allowed Outage Time, Surveillance Test Interval), state, condition, and definition (e.g., operability) should be clearly specified. As an example, a number might be based on engineering judgment, past experience, and/or PRA insights but this should be clearly stated.

The Commission recognizes that certain amendments to the regulations[2] may be necessary before the content of Technical Specifications can be limited entirely to the purpose defined above as embodied in the associated criteria (e.g., §50.36a on Radiological Environmental Technical Specifications would have to be amended before radiological effluent controls can be transferred from the Technical Specifications to other documents). The Staff will initiate in parallel with issuance of this Policy Statement the rule changes necessary to fully implement this Policy Statement.

To give added assurance that the conditions and limitations currently contained in Technical Specifications that will be removed are adequately controlled, the NRC will give increased attention to changes made pursuant to §50.59 and to the administrative control requirements of the Technical Specifications. The NRC is paying closer attention to FSAR updates, and will specifically look for changes which potentially violate §50.59. The Staff is encouraging industry to get the help of the Institute of Nuclear Power Operations (INPO) and the support of the Nuclear Utility Management Resource Committee (NUMARC), in sponsoring activities to encourage the highest quality for utility review of changes including those made pursuant to §50.59. The NRC will work with industry to develop a standard for the conduct of §50.59 reviews. This standard will then be afforded regulatory status (e.g., by a separate policy statement, regulatory guide, or generic letter). In the interim, utilities that choose to file an application to amend their Technical

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[2] Ibid, Enclosure 1, Table 5

Specifications in accordance with this Policy Statement must have in place administrative controls to ensure that changes made pursuant to §50.59 are made only after the bases for the requirement have been clearly established and after review by a multidisciplinary review group made up of responsible, technical supervisory personnel, including onsite operations personnel. In addition, if Technical Specification requirements are relocated to plant procedures, then the revised Technical Specifications must contain administrative controls to ensure that they are appropriately maintained and implemented. The Staff will issue guidance on the appropriate control mechanisms for requirements removed from Technical Specifications (e.g., FSAR amendment, procedures, or other licensee-controlled document) in time for use when the Policy Statement is issued in final form.

The NRC will, consistent with its mission, allocate resources as necessary to implement this Policy Statement.

IV. ENFORCEMENT POLICY

Any changes to a licensees' Technical Specifications to apply this Policy Statement's criteria will be made by the license amendment process prior to implementation. Continued compliance with Technical Specifications and with the commitments contained in other licensee-controlled documents is required by the Commission. Violations and deviations will, as in the past, be subject to the Enforcement Policy in 10 CFR Part 2, Appendix C, (1986).

If a licensee elects to apply these criteria, the requirements of the removed specifications will be relocated to the Final Safety Analysis Report (FSAR) or other licensee controlled documents. Licensees must operate their facilities in conformance with the descriptions of their facilities and procedures in their FSAR unless the change is reviewed and approved in accordance with §50.59. The Commission will take appropriate enforcement action to ensure that licensees comply with FSAR commitments and §50.59. Changes to the provisions of other documents (e.g., QA plan, plant procedures) are subject to the specific requirements for those documents.

Nothing in this Policy Statement shall limit the authority of the NRC to conduct inspections as deemed necessary and to take appropriate enforcement action when regulatory requirements or commitments are not met.

ADDITIONAL VIEWS OF COMMISSIONER ASSELSTINE

Commissioner Asselstine adds the following: I disapprove this interim policy statement. Although I support an effort to bring about improvements in plant Technical Specifications, I believe that this policy statement must be modified in four respects: First, any such policy should contain an explicit statement that the Commission will not entertain changes in testing and surveillance intervals and allowed outage times until licensee maintenance programs are strengthened. Second, I believe the 10 CFR 50.59 review process should be strengthened before licensees are given the flexibility afforded this interim policy. Third, this interim policy weakens the Commission's enforcement options for some important safety requirements now contained in the Technical Specifications. For example, plants licensed since January 1, 1979 (33 full power licenses thus far) are not covered by the requirements of the Commission's fire protection regulations (10 CFR Part 50, Appendix R). Instead, the Technical Specifications and license conditions have been used as the vehicle for establishing enforceable fire protection requirements for the plants licensed since 1978. It appears that this policy statement would allow removing the enforceable fire protection requirements from the Technical Specifications and placing them in a far less enforceable document -- the Final Safety Analysis Report. The February 7, 1986 memorandum from the Acting Director for Operations to the Commissioners (Subject: Test Application of TSIP Technical Specification Selection Criteria) indicates that fire detection instrumentation, fire suppression systems and fire barriers would no longer be covered by the Technical Specifications. As the NRC staff admits, "(T)he NRC's ability to fine a licensee or to seek escalated enforcement action against a licensee who fails to comply with some relocated Technical Specifications is somewhat diminished." This is unacceptable. At a minimum, the Commission should treat failures to meet safety provisions in the Final Safety Analysis Report and other such controlled documents in the same manner as failures to comply with Technical Specifications.

Finally, the February 7, 1986 memorandum indicates that AC and DC power sources would not be covered by Technical Specifications while the plant is in the decay heat removal mode. These power sources are not deemed vital because events in this mode or operation are not "design basis accidents." I find this argument troubling. The significance of the decay heat removal function is described in, for example, the NRC's Office of Analysis and Evaluation of Operational Data report "Decay Heat Removal Problems at U.S. Pressurized Water Reactors" AEOD/C503, December, 1985. I fail to see the wisdom of not addressing power sources in the Technical Specifications while the plant is in the decay heat removal mode. Therefore, I must question the adequacy of the selection criteria for what is and is not to remain in the Technical Specifications.

I would appreciate receiving comments on the above.

Dated at Washington, D.C., this _______ day of __________, 1987.

For the Nuclear Regulatory Commission

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Samuel J. Chilk,
Secretary of the Commission.

52 FR 3788 (February 6, 1987)

[7590-01]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors

AGENCY: Nuclear Regulatory Commission.

ACTION: Interim Policy Statement.

SUMMARY: This statement presents the policy of the Nuclear Regulatory Commission (NRC) with respect to the scope and purpose of Technical Specifications for nuclear power plants as required by 10 CFR 50.36. It establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. It encourages licensees to implement a voluntary program to update their Technical Specifications to be consistent with revised vendor-specific Standard Technical Specifications (STS) to be developed by the industry based on these criteria and subject to NRC Staff approval. The Policy Statement also identifies mechanisms to be used by the NRC and industry to control changes to those items removed from Technical Specifications. The Policy Statement is expected to produce an improvement in the safety of nuclear power plants through the development of more operator-oriented Technical Specifications, improved Technical Specification Bases, reduced action statement-induced plant transients, and more efficient use of NRC and industry resources.

DATE: This Interim Policy Statement is effective upon issuance. However, the public is invited to submit comments by March 23, 1987. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before this date. On the basis of the submitted comments, the Commission will determine whether to modify the Policy Statement before issuing it as final.

FOR FURTHER INFORMATION CONTACT: David C. Fischer, Technical Specifications Coordination Branch, Division of Human Factors Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, telephone (301) 492-7924.

SUPPLEMENTARY INFORMATION:

I. BACKGROUND

Section 182a. of the Atomic Energy Act of 1954, as amended (42 U.S.C. 2232), mandates the inclusion of Technical Specifications in licenses for the operation of production and utilization facilities. The Act requires that Technical Specifications include information of the amount, kind, and source of special nuclear material, the place of use, and the specific characteristics of the facility. That section also indicates that Technical Specifications should contain such information as the Commission may by rule deem necessary to enable it to find that the utilization of special nuclear material will be in accord with the common defense and will provide adequate protection of public health and safety. Finally, that section requires Technical Specifications to be made a part of any license issued.

Section 50.36, "Technical Specifications," which implements Section 182a. of the Atomic Energy Act, was promulgated by the Commission on December 17, 1968 (33 FR 18610). This rule delineates requirements for determining the contents of Technical Specifications. Technical Specifications set forth the specific characteristics of the facility and the conditions for its operation that are required to provide adequate protection to the health and safety of the public. Specifically, 10 CFR 50.36 requires that:

"Each license authorizing operation of a production or utilization facility of a type described in §50.21 or §50.22 will include Technical Specifications. The Technical Specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to §50.34. The Commission may include such additional Technical Specifications as the Commission finds appropriate."

Technical Specifications cannot be changed by licensees without prior NRC approval. However, since 1969, there has been a trend towards including in Technical Specifications not only those requirements derived from the analyses and evaluation included in the safety analysis report but also essentially all other Commission requirements governing the operation of nuclear power reactors. This extensive use of Technical Specifications is due in part to a lack of well defined criteria (in either the body of the rule or in some other regulatory document) for what should be included in Technical Specifications. This has contributed to the volume of Technical Specifications and to the several fold increase, since 1969, in the number of license amendment applications to effect changes to the Technical Specifications. It has diverted both staff and licensee attention from the more important requirements in these documents to the extent that it has resulted in an adverse but unquantifiable impact on safety.

On March 30, 1982, the NRC published in the Federal Register (47 FR 13369) a proposed amendment to its regulations, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The proposed amendment would have revised §50.36, "Technical Specifications," to establish a new system of specifications divided into two general categories. Only those specifications contained in the first general category as Technical Specifications would have become part of the operating license and require prior NRC approval for any changes. Those specifications contained in the second general category would have become supplemental specifications and would not require prior NRC approval for most changes. The NRC review of the first general category of specifications would have been the same as currently performed for Technical Specifications changes, which are amendments to the operating license. For the second category, supplemental specifications, the licensee would have been allowed to make changes within specified conditions without prior NRC approval. The NRC would have reviewed these changes when they were made and would have done so in a manner similar to that currently used for reviewing design changes, tests, and experiments performed under the provisions of 10 CFR 50.59.

Because of difficulties with defining the criteria for dividing the Technical Specifications into the two categories of the proposed rule and other higher priority licensing work, the rule change was deferred.

In the past several years the nuclear industry and the NRC Staff have been studying the question of whether improvement to the current system of establishing Technical Specification requirements for nuclear power plants is needed. The two most recent studies of this issue were performed by an NRC task group known as the Technical Specifications Improvement Project (TSIP) and a Subcommittee of the Atomic Industrial Forum's (AIF) Committee on Reactor Licensing and Safety.[1] The overall conclusion of these studies was that many improvements in the scope and content of Technical Specifications are needed, and that a joint NRC and Industry program should be initiated to implement these improvements. Both of these groups made specific recommendations which are summarized as follows:

1) The NRC should adopt the criteria for defining the scope of Technical Specifications proposed in the AIF and TSIP reports. Those criteria should then be used by the NRC and each of the nuclear steam supply

[1] SECY-86-10, "Recommendations for Improving Technical Specification," dated January 13, 1986, contains both "Recommendations for Improving Technical Specifications," NRC Technical Specifications Improvement Project, September 30, 1985, and "Technical Specifications Improvements," AIF Subcommittee on Technical Specifications Improvements, October 1, 1985.

system vendor owners groups to completely rewrite and streamline the existing Standard Technical Specifications (STS). This process would result in many requirements being transferred from control by Technical Specification requirements to control by other mechanisms [e.g., the Final Safety Analysis Report (FSAR), Operating Procedures, Quality Assurance (QA) Plan] which would not require a license amendment or prior NRC approval when changes are needed. The new STS should include greater emphasis on human factors principles in order to add clarity and understanding to the text of the STS. The new STS should also provide improvements to the Bases Section of Technical Specifications which provides the purpose for each requirement in the specification.

2) A parallel program of short-term improvements in both the scope and substance of the existing Technical Specifications should be initiated in addition to developing a new STS as identified in (1) above.

II. DISCUSSION

The Commission recognizes the advantages of improved Technical Specifications. Clarification of the scope and purpose of Technical Specifications will provide useful guidance to both the NRC and industry and should serve as an important incentive for industry participation in a voluntary program to improve Technical Specifications. It will result in Technical Specifications that focus licensee's and the plant operator's attention on those plant conditions most important to safety and should also result in more efficient use of agency and industry resources.

The Policy Statement identifies three objective criteria for defining the scope of Technical Specifications. These criteria are intended to be consistent with the scope of Technical Specifications as stated in the Statement of Consideration accompanying the current rule.

The Statement of Consideration discusses the scope of Technical Specifications as including the following:

> "In the revised system, emphasis is placed on two general classes of technical matters: (1) those related to prevention of accidents, and (2) those related to mitigation of the consequences of accidents. By systematic analysis and evaluation of a particular facility, each applicant is required to identify at the construction permit stage, those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity. Such items are expected to be the subjects of Technical Specifications in the operating license."

33 FR 18610 (December 17, 1968). The first of these two general classes of technical matters to be included in Technical Specifications is captured by criterion (1) and to some extent criterion (2) in that they address systems and process variables that alert the operator to a situation when accident initiation is more likely. The second general class of technical matters is explicitly addressed and captured by criteria (2) and (3). By applying the three criteria contained in the Policy Statement a licensee should capture all of those specific characteristics of its facility and the conditions for its operation that are required to meet the principal operative standard in Section 182a. of the Atomic Energy Act, that is, that adequate protection is provided to the health and safety of the public.

The Commission recognizes that the three criteria carry with them a common theme of focusing on those requirements related to technical matters dealing with those features of a facility that are of controlling importance to safety. Since many of the requirements are of immediate concern to the health and safety of the public, the Policy Statement adopts, for the purpose of relocating requirements from Technical Specifications to other licensee-controlled documents, the subjective statement of the purpose of Technical Specifications expressed by an Atomic Safety and Licensing Appeal Board Portland General Electric Company (Trojan Nuclear Plant), ALAB-531, 9 NRC 263 (1979). There the Appeal Board interpreted Technical Specifications as being reserved for those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. The Commission wishes to emphasize that this Policy Statement is intended to be consistent with the language of Section 182a. of the Atomic Energy Act, 10 CFR 50.36, and previous interpretations of the regulations. It merely clarifies the scope and purpose of Technical Specifications by identifying criteria which can be used to establish, more clearly, the framework for Technical Specifications (i.e., identify those requirements derived from the analyses and evaluation included in the safety analysis report and which are of immediate concern to the health and safety of the public). It identifies requirements which should be retained in Technical Specifications and also describes a mechanism whereby other "additional" requirements can be identified and controlled through mechanisms other than Technical Specifications.

The Commission invites public comment on this Policy Statement and particularly invites comment on the statement of the purpose of Technical Specifications which introduces the text of the Policy Statement and on whether it would be beneficial for licensees to be able to modify related portions of their LCOs (such as containment systems) without having to apply the terms and provisions of the Policy Statement to all LCOs.

III. THE COMMISSION'S POLICY

The purpose of Technical Specifications is to impose those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety by establishing those conditions of operation which cannot be changed without prior Commission approval and by identifying those features which are of controlling importance to safety.

Licensees are encouraged to implement a program to upgrade their Technical Specifications consistent with this purpose. The Commission will entertain requests based on the criteria below (as clarified by the supporting discussion) for individual license amendments that evaluate all of the Limiting Conditions for Operation (LCOs) for an individual plant to determine which LCOs should be included in the Technical Specifications. The Commission does not intend that these criteria be used as the basis for relocation of individual LCOs. LCOs which fail to meet any one or more of the criteria below may be removed from the Technical Specifications and relocated to other licensee-controlled documents, such as the FSAR or licensee procedures. The criteria may be applied to either Standard or custom Technical Specifications. However, it is expected that each of the nuclear steam supply system vendor owners groups will undertake the development of revised STS based on this Policy Statement, and we encourage licensees to use the revised STS as the basis for their individual plant Technical Specifications. The NRC will give first priority in its Technical Specifications improvements efforts to the review and approval of the revised STS and the plant specific license amendment applications based on them. Approved short term Technical Specifications improvements will be included in the revised STS. The revised STS and individual license amendment requests that are submitted based on this Policy Statement should incorporate all terms and provisions of the Policy Statement.

The following criteria delineate those constraints on design and operation of nuclear power plants that are derived from the plant safety analysis report and belong in Technical Specifications in accord with 10 CFR 50.36 and the purpose of Technical Specifications stated above.

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary:

> Discussion of Criterion 1: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident.

This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage.

Criterion 2: A process variable that is an initial condition of a Design Basis Accident (DBA) or Transient Analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 2: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing Design Basis Accident and Transient Analyses. These analyses consist of postulated events, analyzed in the Final Safety Analysis Report (FSAR), for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the FSAR (or equivalent chapters) and are identified as Condition II, III, or IV events (ANSI N 18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

As used in Criterion 2, process variables are only those parameters for which specific values or ranges of values have been chosen as reference bounds in the Design Basis Accident or Transient Analyses and which are monitored and controlled during power operation such that process values remain within the analysis bounds.

The purpose of this criterion is to capture those process variables that have initial values assumed in the Design Basis Accident and Transient Analyses, and which are monitored and controlled during power operation. So long as these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated Design Basis Accident or Transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequence of the Design Basis Accident or Transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's Design Basis Accident and Transient Analyses, as presented in Chapters 6 and 15 of the plant's Final Safety Analysis Report (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to Design Basis Accidents and Transients limits the consequences of these events to within the appropriate acceptance criteria.

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function.

In addition to those structures, systems, and components captured by the above criteria, it is the Commission's policy that licensees retain in their Technical Specifications LCOs, action statements, and Surveillance Requirements for the following systems (as applicable) which operating experience and probabilistic risk assessment have generally shown to be important to public health and safety:

- Reactor Core Isolation Cooling (RCIC)/Isolation Condenser,
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- Recirculation Pump Trip (RPT).

The Commission recognizes that features of plant design and operation not addressed in the safety analysis report's Design Basis Accidents or Transient Analyses can, in some cases, be significant contributors to the plant's overall core melt probability and risk. As stated in 10 CFR 50.36, the Commission may include such additional Technical Specifications as the Commission finds appropriate. Based on this, and consistent with the Commission's Safety Goal and Severe Accident Policy Statements, the Commission finds that risk evaluations are an appropriate tool for defining requirements that should be retained in Technical Specifications where including such requirements is consistent with the purpose of Technical Specifications as defined above.

The Commission expects that owners groups, in preparing their proposals to streamline the Standard Technical Specifications, will utilize the available literature on risk insights and Probabilistic Risk Assessments (PRAs). This material should be employed to strengthen the technical bases for those requirements that remain in Technical Specifications, when applicable, and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk. Similarly, the Staff will also employ risk insights and PRAs in evaluating the revised STS.

In some cases, plant-specific PRAs or risk surveys conducted, for example, pursuant to the Commission's Severe Accident Policy, may be available to licensees as they prepare license amendments to adopt the revised STS to their plant, or to streamline custom Technical Specifications under this Policy Statement. Where such PRAs or surveys are available, they should be used to strengthen the Bases and screen those Technical Specifications to be relocated, as suggested above. Where such plant-specific risk surveys are unavailable, licensees should utilize the available literature on risk insights and PRAs, as described above. However, licensees need not await the performance of plant-specific PRA studies before availing themselves of this policy. As in the case of the revised STS discussed above, the Staff will also utilize risk insights and PRAs in evaluating the plant-specific submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue research in methods to make better use of risk and reliability considerations for defining future generic Technical Specification requirements.

Requirement(s) which would be relocated from Technical Specifications to another licensee-controlled document (e.g., the FSAR and 10 CFR 50.59, Operating Procedures, the QA Plan, or Fire Protection Plan) may be changed or deleted in conjunction with the filing of the revised STS or of individual license amendment request to implement this Policy Statement. The package containing the revised STS or the amendment request must contain a clear statement of the basis of the requirement(s) to be changed or deleted, a safety evaluation, and a statement that the change(s) has been reviewed by a multidisciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

When licensees submit amendment requests based on this Policy Statement, they should identify the location of, and controls for, the technical and administrative requirements of the removed Technical Specifications. The Staff will carefully review these submittals to ensure the accountability of each relocated requirement.

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1. What is the justification for the Technical Specification, i.e., which criterion requires it to be in the Technical Specifications?

2. What are the Bases for each Limiting Condition for Operation (LCO), i.e., why was it determined to be the lowest functional capability or performance level for the system/component in question necessary for safe operation of the facility and what are the reasons for the Applicable Operational Modes(s) for the LCO?

3. What are the Bases for each Action Statement, i.e., why should this remedial action be taken if the associated LCO cannot be met, how does this action relate to other Action Statements associated with the LCO, and what justifies continued operation of the system/component at the reduced state from the state specified in the LCO for the allowed time period?

4. What are the Bases for each Limiting Safety System Setting?

5. What are the Bases for each Surveillance Requirement and the surveillance interval specified, i.e., what specific functional requirement is the surveillance designed to verify, and why is this surveillance necessary at the specified frequency to assure that the system/component function is maintained, that facility operation will be within the safety limits, and that the LCO will be met?

NOTE: In answering these questions the Bases for each number (e.g., Trip Set point, Response Time, Allowed Outage Time, Surveillance Test Interval), state, condition, and definition (e.g., operability) should be clearly specified. As an example, a number might be based on engineering judgment, past experience, and/or PRA insights but this should be clearly stated.

The Commission recognizes that certain amendments to the regulations[2] may be necessary before the content of Technical Specifications can be limited entirely to the purpose defined above as embodied in the associated criteria (e.g., §50.36a on Radiological Environmental Technical Specifications would have to be amended before radiological effluent controls can be transferred from the Technical Specifications to other documents). The Staff will initiate in parallel with issuance of this Policy Statement the rule changes necessary to fully implement this Policy Statement.

To give added assurance that the conditions and limitations currently contained in Technical Specifications that will be removed are adequately controlled, the NRC will give increased attention to changes made pursuant to §50.59 and to the administrative control requirements of the Technical Specifications. The NRC is paying closer attention to FSAR updates, and will specifically look for changes which potentially violate §50.59. The Staff is encouraging industry to get the help of the Institute of Nuclear Power Operations (INPO) and the support of the Nuclear Utility Management Resource Committee (NUMARC), in sponsoring activities to encourage the highest quality for utility review of changes including those made pursuant to §50.59. The NRC will work with industry to develop a standard for the conduct of §50.59 reviews. This standard will then be afforded regulatory status (e.g., by a separate policy statement, regulatory guide, or generic letter). In the interim, utilities that choose to file an application to amend their Technical

---

[2]Ibid, Enclosure 1, Table 3

Specifications in accordance with this Policy Statement must have in place administrative controls to ensure that changes made pursuant to §50.59 are made only after the bases for the requirement have been clearly established and after review by a multidisciplinary review group made up of responsible, technical supervisory personnel, including onsite operations personnel. In addition, if Technical Specification requirements are relocated to plant procedures, then the revised Technical Specifications must contain administrative controls to ensure that they are appropriately maintained and implemented. The Staff will issue guidance on the appropriate control mechanisms for requirements removed from Technical Specifications (e.g., FSAR amendment, procedures, or other licensee-controlled document) in time for use when the Policy Statement is issued in final form.

The NRC will, consistent with its mission, allocate resources as necessary to implement this Policy Statement.

IV. ENFORCEMENT POLICY

Any changes to a licensees' Technical Specifications to apply this Policy Statement's criteria will be made by the license amendment process prior to implementation. Continued compliance with Technical Specifications and with the commitments contained in other licensee-controlled documents is required by the Commission. Violations and deviations will, as in the past, be subject to the Enforcement Policy in 10 CFR Part 2, Appendix C, (1986).

If a licensee elects to apply these criteria, the requirements of the removed specifications will be relocated to the Final Safety Analysis Report (FSAR) or other licensee controlled documents. Licensees must operate their facilities in conformance with the descriptions of their facilities and procedures in their FSAR unless the change is reviewed and approved in accordance with §50.59. The Commission will take appropriate enforcement action to ensure that licensees comply with FSAR commitments and §50.59. Changes to the provisions of other documents (e.g., QA plan, plant procedures) are subject to the specific requirements for those documents.

Nothing in this Policy Statement shall limit the authority of the NRC to conduct inspections as deemed necessary and to take appropriate enforcement action when regulatory requirements or commitments are not met.

ADDITIONAL VIEWS OF COMMISSIONER ASSELSTINE

Commissioner Asselstine adds the following: I disapprove this interim policy statement. Although I support an effort to bring about improvements in plant Technical Specifications, I believe that this policy statement must be modified in four respects: First, any such policy should contain an explicit statement that the Commission will not entertain changes in testing and surveillance intervals and allowed outage times until licensee maintenance programs are strengthened. Second, I believe the 10 CFR 50.59 review process should be strengthened before licensees are given the flexibility afforded this interim policy. Third, this interim policy weakens the Commission's enforcement options for some important safety requirements now contained in the Technical Specifications. For example, plants licensed since January 1, 1979 (33 full power licenses thus far) are not covered by the requirements of the Commission's fire protection regulations (10 CFR Part 50, Appendix R). Instead, the Technical Specifications and license conditions have been used as the vehicle for establishing enforceable fire protection requirements for the plants licensed since 1978. It appears that this policy statement would allow removing the enforceable fire protection requirements from the Technical Specifications and placing them in a far less enforceable document -- the Final Safety Analysis Report. The February 7, 1986 memorandum from the Acting Director for Operations to the Commissioners (Subject: Test Application of TSIP Technical Specification Selection Criteria) indicates that fire detection instrumentation, fire suppression systems and fire barriers would no longer be covered by the Technical Specifications. As the NRC staff admits, "(T)he NRC's ability to fine a licensee or to seek escalated enforcement action against a licensee who fails to comply with some relocated Technical Specifications is somewhat diminished." This is unacceptable. At a minimum, the Commission should treat failures to meet safety provisions in the Final Safety Analysis Report and other such controlled documents in the same manner as failures to comply with Technical Specifications.

Finally, the February 7, 1986 memorandum indicates that AC and DC power sources would not be covered by Technical Specifications while the plant is in the decay heat removal mode. These power sources are not deemed vital because events in this mode or operation are not "design basis accidents." I find this argument troubling. The significance of the decay heat removal function is described in, for example, the NRC's Office of Analysis and Evaluation of Operational Data report "Decay Heat Removal Problems at U.S. Pressurized Water Reactors" AEOD/C503, December, 1985. I fail to see the wisdom of not addressing power sources in the Technical Specifications while the plant is in the decay heat removal mode. Therefore, I must question the adequacy of the selection criteria for what is and is not to remain in the Technical Specifications.

I would appreciate receiving comments on the above.

Dated at Washington, D.C., this _______ day of __________, 1987.

For the Nuclear Regulatory Commission

____

Samuel J. Chilk,
Secretary of the Commission.

December 4, 1990

MEMORANDUM FOR: Edward L. Jordan, Director
Office for Analysis and Evaluation of
Operational Data

FROM: Frank J. Miraglia, Jr., Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: CRGR BRIEFING ON THE NEW STANDARD TECHNICAL SPECIFICATIONS (STS)

NRR is scheduled to brief CRGR on the new Standard Technical Specifications on December 12, 1990. It is anticipated that a final draft of the new STS will be issued to the owners groups for comment in the very near future. It is not necessary to have reviewed the new STS prior to the briefing since this briefing is intended only to introduce the new STS to CRGR. It is anticipated that future meetings will be scheduled at which the major issues can be discussed in detail, if desired.

In order to provide some background information for the first briefing, we are providing the following documents to CRGR members and staff:

1. Commission (interim) Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

2. Letters to the owners group chairmen providing lists of requirements which may be relocated from the STS, May 9, 1988.

3. SECY-88-304 Staff Actions to Reduce Testing at Power, October 26, 1988.

4. SECY-90-366 Report on the Status of the Technical Specifications Improvement Program, October 29, 1990.

The contact for this effort is Mr. Richard Lobel (x21185). This effort is sponsored by Charles E. Rossi, Director, Division of Operational Events Assessment.

We look forward to introducing CRGR to the large amount of work which has been done by the staff and the industry to improve the technical specifications.

Original signed by
Frank J. Miraglia

Frank J. Miraglia, Jr., Deputy Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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ENCLOSURE 2

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

MAY 9 1988

Mr. R. A. Newton, Chairman
Westinghouse Owners Group
Wisconsin Electric Power Company
P.O. Box 2046
Milwaukee, WI 53201

Dear Mr. Newton:

This letter is in response to your report identifying which Standard Technical Specification (STS) requirements you believe should be retained in the new STS and which can be relocated to other licensee-controlled documents.

The enclosure to this letter documents the NRC staff's conclusions as to which current STS requirements must be retained in the new STS. These conclusions are based on the Commission's Interim Policy Statement on Technical Specification Improvements and on several interpretations of how to apply the screening criteria contained in that Policy Statement. The NRC staff considered comments made by industry at a March 29, 1988 meeting between NRC, NUMARC, and each Owners Group in making these interpretations.

Based on our review, we have concluded that a significant reduction can be made in the number of Limiting Conditions for Operation (and associated Surveillance Requirements) that must be included in the STS. Our goal is to assure that the new STS contain only requirements that are consistent with 10 CFR 50.36 and have a sound safety basis.

The development of the new STS based on the staff's conclusions will result in more efficient use of NRC and industry resources. Safety improvements are expected through more operator-oriented Technical Specifications, improved Technical Specification Bases, a reduction in action statement-induced plant transients, and a reduction in testing at power.

As you are aware, the NRC staff and industry also have underway a parallel program of specific line item improvements to both the scope and substance of the existing Technical Specifications. The need for many of these types of improvements was identified in the report (NUREG-1024) of a major staff task group established in 1983 to study surveillance requirements in Technical Specifications and develop alternative approaches to provide better assurance that surveillance testing does not adversely impact safety. The NRC will continue to actively identify and pursue the development of specific line item improvements to Technical Specifications and will make these improvements immediately available to licensees without waiting for the new STS. We encourage each of the Owners Groups to continue to work with the NRC staff on these types of parallel improvements to existing Technical Specifications.

We are confident that the enclosed staff report provides an adequate basis for the Owners Groups to proceed with the development of complete new STS in accordance with the Commission's Interim Policy Statement.

We will continue to interact with the NUMARC Technical Specification Working Group and each of the individual vendor Owners Groups as needed to keep this important program moving forward.

Sincerely,

TEMurley

Thomas E. Murley, Director
Office of Nuclear ~~Reactor~~ Regulation

Enclosure:
As stated

cc see next page

cc w/encl:

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

MAY 9 1988

Mr. Walter S. Wilgus, Chairman
The B&W Owners Group
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852

Dear Mr. Wilgus:

This letter is in response to your report identifying which Standard Technical Specification (STS) requirements you believe should be retained in the new STS and which can be relocated to other licensee-controlled documents.

The enclosure to this letter documents the NRC staff's conclusions as to which current STS requirements must be retained in the new STS. These conclusions are based on the Commission's Interim Policy Statement on Technical Specification Improvements and on several interpretations of how to apply the screening criteria contained in that Policy Statement. The NRC staff considered comments made by industry at a March 29, 1988 meeting between NRC, NUMARC, and each Owners Group in making these interpretations.

Based on our review, we have concluded that a significant reduction can be made in the number of Limiting Conditions for Operation (and associated Surveillance Requirements) that must be included in the STS. Our goal is to assure that the new STS contain only requirements that are consistent with 10 CFR 50.36 and have a sound safety basis.

The development of the new STS based on the staff's conclusions will result in more efficient use of NRC and industry resources. Safety improvements are expected through more operator-oriented Technical Specifications, improved Technical Specification Bases, a reduction in action statement-induced plant transients, and a reduction in testing at power.

As you are aware, the NRC staff and industry also have underway a parallel program of specific line item improvements to both the scope and substance of the existing Technical Specifications. The need for many of these types of improvements was identified in the report (NUREG-1024) of a major staff task group established in 1983 to study surveillance requirements in Technical Specifications and develop alternative approaches to provide better assurance that surveillance testing does not adversely impact safety. The NRC will continue to actively identify and pursue the development of specific line item improvements to Technical Specifications and will make these improvements immediately available to licensees without waiting for the new STS. We encourage each of the Owners Groups to continue to work with the NRC staff on these types of parallel improvements to existing Technical Specifications.

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We are confident that the enclosed staff report provides an adequate basis for the Owners Groups to proceed with the development of complete new STS in accordance with the Commission's Interim Policy Statement.

We will continue to interact with the NUMARC Technical Specification Working Group and each of the individual vendor Owners Groups as needed to keep this important program moving forward.

Sincerely,

Original signed by
Thomas E. Murley
Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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GCLainas
FSchroeder
JRichardson

(W.S.WILGUS/LTR/SPLIT REPORT)

CONCURRENCE:

*(see previous concurrence)

*TSB:DOEA:NRR	*TSB:NRR	*C:TSB:NRR	*D:DOEA:NRR	*D:DEST:NRR	*D:DEST:NRR
KDesai:pmc	DCFischer	EJButcher	CERossi	AThadani	LShao
4/18/88	04/19/88	04/20/88	04/22/88	04/26/88	04/26/88

*D:DREP:NRR	*ADT:NRR	D:NRR
JRStohr	TTMartin	TEMurley
04/28/88	05/05/88	05/6/88

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Dr. J. K. Gasper, Chairman
CE Owners Group
Omaha Public Power District
1623 Harney Street
ATTN: Jones St. Station
Omaha, Nebraska 68102

Mr. Robert F. Janecek, Chairman
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c/o Commonwealth Edison Company
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Chicago, IL 60690

NRC STAFF REVIEW

OF

NUCLEAR STEAM SUPPLY SYSTEM VENDOR OWNERS GROUPS'

APPLICATION OF

THE COMMISSION'S INTERIM POLICY STATEMENT CRITERIA

TO

STANDARD TECHNICAL SPECIFICATIONS

1. INTRODUCTION

On February 6, 1987, the Commission issued its Interim Policy Statement on Technical Specification Improvements (52 FR 3788). The Policy Statement encourages the industry to develop new Standard Technical Specifications (STS) to be used as guides for licensees in preparing improved Technical Specifications (TS) for their facilities. The Interim Policy Statement contains criteria (including a discussion of each) for determining which regulatory requirements and operating restrictions should be retained in the new STS and ultimately in plant TS. It also identifies four additional systems that are to be retained on the basis of operating experience and probabilistic risk assessments (PRA). Finally, the Policy Statement indicates that risk evaluations are an appropriate tool for defining requirements that should be retained in the STS/TS where including such requirements is consistent with the purpose of TS (as stated in the Policy Statement). Requirements that are not retained in the new STS would generally not be retained in individual plant TS. Current TS requirements not retained in the STS will be relocated to other licensee-controlled documents.

One of the first steps in the program to implement the Commission's Interim Policy Statement is to determine which Limiting Conditions for Operation (LCOs) contained in the existing STS should be retained in the new STS. An early decision on this issue will facilitate efforts to make the other improvements (described in the Policy Statement) to the text and Bases of those requirements that must be retained in the new STS.

Each Nuclear Steam Supply System (NSSS) vendor Owners Group has submitted a report to the NRC for review that identifies which STS LCOs the group believes should be retained in the new STS and which can be relocated to other licensee-controlled documents. These four NSSS vendor submittals are as follows:

(1) Letter dated October 15, 1987, R. L. Gill, B&W Owners Group, to Dr. T. E. Murley, NRC, Subject: "B&W Owners Group Technical Specification Committee Application of Selection Criteria to the B&W Standard Technical Specifications."

(2) Letter dated November 12, 1987, R. A. Newton, Westinghouse Owners Group, to NRC Document Control Desk, Subject: "Westinghouse Owners Group MERITS Program Phase II, Task 5, Criteria Application Topical Report."

(3) Letter dated December 11, 1987, J. K. Gasper, Combustion Engineering Owners Group, to Dr. T. E. Murley, NRC Subject: "CEN-355, CE Owners Group Restructured Standard Technical Specifications - Volume 1 (Criteria Application)."

(4) Letter dated November 12, 1987, R. F. Janecek, BWR Owners Group, to R. E. Starostecki, NRC, Subject: "BWR Owners Group Technical Specification screening Criteria Application and Risk Assessment."

These submittals provide the rationale for why each STS requirement (e.g. Limiting Condition for Operation) should be retained in the new STS or why it can be relocated to a licensee-controlled document. They also describe how each Owners Group used risk insights in determining the appropriate content of the new STS.

2. STAFF REVIEW

The NRC staff focused its review on those requirements identified by the Owners Groups as candidates for relocation. The staff evaluated each of these requirements to determine whether it agreed with the Owners Groups' conclusions.

During the NRC Staff's review, several issues were raised concerning the proper interpretation or application of the criteria in the Commission's Interim Policy Statement. The NRC Staff has considered these issues and concluded the following:

(1) Criterion 1 should be interpreted to include only instrumentation used to detect actual leaks and not more broadly to include instrumentation used

to detect precursors to an actual breech of the reactor coolant pressure boundary or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).

(2) The "initial conditions" captured under Criterion 2 should not be limited to only "process variables" assumed in safety analyses. They should also include certain active design features (e.g., high pressure/low pressure system valves and interlocks) and operating restrictions (e.g., pressure-temperature operating limit curves), needed to preclude unanalyzed accidents. In this context, "active design features" include only design features under the control of operations personnel (i.e., licensed operators and personnel who perform control functions at the direction of licensed operators). This position is consistent with the conclusions reached by the Staff during the trial application of the criteria to the Wolf Creek and Limerick Technical Specifications.

(3) The "initial conditions" of design-basis accidents (DBA) and transients, as used in Criterion 2, should not be limited to only those directly "monitored and controlled" from the control room. Initial conditions should also include other features/characteristics that are specifically assumed in DBA and transient analyses even if they can not be directly observed in the control room. For example, initial conditions (e.g., moderator temperature coefficient and hot channel factors) that are periodically monitored by other than licensed operators (e.g., core engineers, instrumentation and control technicians) to provide licensed operators with the information required to take those actions necessary to assure that the plant is being operated within the bounds of design and analysis assumptions, meet Criterion 2 and should be retained in Technical Specifications. Initial conditions do not, however, include things that are purely design requirements.

(4) The phrase "primary success path," used in Criterion 3, should be interpreted to include only the primary equipment (including redundant trains/components) to mitigate accidents and transients. Primary success path does not include backup and diverse equipment or instrumentation used to prevent analyzed

accidents or transients or to improve reliability of the mitigation function (e.g., rod withdrawal block which is backup to the average power range monitor high flux trip in the startup mode, safety valves which are backup to low temperature over pressure relief valves during cold shutdown).

(5) Post-Accident Monitoring Instrumentation that satisfies the definition of Type A variables in Regulatory Guide 1.97, "Instrumentation for Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," meets Criterion 3 and should be retained in Technical Specifications. Type A variables provide primary information (i.e., information that is essential for the direct accomplishment of the specified manual actions (including long-term recovery actions) for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs or transients). Type A variables do not include those variables associated with contingency actions that may also be identified in written procedures to compensate for failures of primary equipment. Because only Type A variables meet Criterion 3, the STS should contain a narrative statement that indicates that individual plant Technical Specifications should contain a list of Post-Accident Instrumentation that includes Type A variables. Other Post-Accident Instrumentation (i.e., non-Type A Category I) is discussed on page 6.

(6) The NRC's design basis for licensing a plant is the plant's Final Safety Analysis Report (FSAR) as qualified by the analysis performed by the staff and documented in the staff's safety evaluation report (SER). Because the staff's review and resulting SER are based on the acceptance criteria in the NRC's Standard Review Plan (NUREG-0800, SRP), the dose limits used in licensing a particular plant may be "some small fraction" of those specified in the Commission's regulations in Title 10 of the Code of Federal Regulations Part 100 (10 CFR 100). Accordingly, the SRP limits should be used to define the equipment in the primary success path for mitigating accidents and transients when developing the new STS. These types of conservatisms are required to compensate for uncertainties in analysis techniques and

provide reasonable assurance that the absolute numerical limits of the regulations will be satisfied.

On a plant-specific basis, systems and equipment that are identified in the NRC staff SER and assumed by the staff to function are considered part of the licensing basis for the plant and are captured by Criterion 3 (e.g., radiation monitoring instrumentation that initiates an isolation function, penetration room exhaust air cleanup system).

(7) DBA and transients, as used in Criteria 2 and 3, should be interpreted to include any design-basis event described in the FSAR (i.e., not just those events described in Chapters 6 and 15 of the FSAR). For example, there may be requirements for some plants which should be retained in Technical Specifications because of the risks associated with some site-specific characteristic (e.g., although not normally required, a Technical Specification on the chlorine detection system might be appropriate where a significant chlorine hazard exists in the site vicinity; similarly, a Technical Specification on flood protection might be appropriate where a plant is particularly vulnerable to flooding and is designed with special flood protection features). Criteria 2 and 3 should not be interpreted to include purely generic design requirements applicable to all plants (e.g., the requirements of General Design Criterion 19 in Appendix A to 10 CFR Part 50 for control room design).

The NRC staff has used the Commission's Interim Policy Statement and the conclusions described above to define the appropriate content of the new STS. The staff plans to factor these conclusions into the Final Policy Statement on Technical Specification Improvements that will be proposed to the Commission.

The staff reviewed the methodology and results provided by each Owners Group to verify that none of the requirements proposed for relocation contains constraints of prime importance in limiting the likelihood or severity of accident sequences that are commonly found to dominate risk. For the purpose of this application of the guidance in the Commission Policy Statement, the staff agrees with the Owners Groups' conclusions except in two areas. First, the staff finds that the Remote Shutdown Instrumentation meets the Policy Statement criteria for inclusion in Technical Specifications based on risk; and second, the staff is unable to confirm the Owners Groups' conclusion that Category 1 Post-Accident Monitoring Instrumentation is not of prime importance in limiting risk. Recent PRAs have shown the risk significance of operator recovery actions which would require a knowledge of Category 1 variables. Furthermore, recent severe accident studies have shown significant potential for risk reduction from accident management. The Owners Groups' should develop further risk-based justification in support of relocating any or all Category 1 variables from the Standard Technical Specifications.

As stated in the Commission's Interim Policy Statement, licensees should also use plant-specific PRAs or risk surveys as they prepare license amendments to adopt the revised STS to their plant. Where PRAs or surveys are available, licensees should use them to strengthen the Bases as well as to screen those Technical Specifications to be relocated. Where such plant-specific risk surveys are not available, licensees should use the literature available on risk insights and PRAs. Licensees need not complete a plant-specific PRA before they can adopt the new STS. The NRC staff will also use risk insights and PRAs in evaluating the plant-specific submittals.

3. RESULTS OF THE STAFF'S REVIEW

Appendices A through D present the detailed results of the staff's review of the Babcock and Wilcox, Westinghouse, Combustion Engineering, and General Electric application of the selection criteria to the existing STS. Each Appendix consists of two tables. Table 1 identifies those LCOs that must be retained in the new STS. Table 2 lists those LCOs that may be wholly or partially relocated to licensee-controlled documents (or be reformatted as a surveillance requirement for another LCO). Where the staff placed specific conditions on relocation of particular LCOs the staff has so noted in the Tables. As a part of the plant specific implementation of the new STS, the staff plans to review the location of, and controls over, relocated requirements. In as much as practicable, the Owners Groups should propose standard locations for, and controls over, relocated requirements.

For each LCO listed in Table 1, the criterion (criteria) that required that the LCO be retained in Technical Specifications is identified. If an LCO was retained in Technical Specifications solely on the basis of risk, "Risk" appears in the criteria column. Where an Owners Group determined that an LCO had to stay in Technical Specifications (because of either a particular criterion or risk) and the Staff agreed that the LCO should be retained in Technical Specifications, the staff did not, in general, verify the Owners Group's basis for retention. However, in several instances the Owners Groups cited risk considerations alone as the basis for retaining Technical Specifications and the staff disagreed with the Owners Groups. In these instances, the staff's basis for retention appears in the criteria column of Table 1.

Any LCO not specifically identified in Table 1 or Table 2 (e.g., an LCO unique to an STS not addressed in the Owners Groups submittals such as the BWR5 STS) should be retained in the STS until the Owners Group proposes and the staff makes a specific determination that it can be relocated to a licensee-controlled document.

Notwithstanding the results of this review, the staff will give further consideration for relocation of additional LCOs as the staff and industry proceed with the development of the new STS.

4. CONCLUSION

The results of the effort of the Owners Groups and of the NRC staff to apply the Policy Statement selection criteria to the existing STS are an important step toward ensuring that the new STS contain only those requirements that are consistent with 10 CFR 50.36 and have a sound safety basis. As shown in the following tables, application of the criteria contained in the Commission's Interim Policy Statement resulted in a significant reduction in the number of LCOs to be included in the new STS. The development of the new STS based on the staff's conclusions will result in more efficient use of NRC and industry resources. Safety improvements are expected through more operator-oriented Technical Specifications, improved Technical Specification Bases, a reduction in action statement-induced plant transients, and a reduction in testing at power.

LCOs	BABCOCK & WILCOX	WESTINGHOUSE	COMBUSTION ENGINEERING	GENERAL ELECTRIC BWR4/BWR6
Total Number	137	165	159	124/144
Retained	75	92	87	81/86
Relocated	62	73	72	43/58
Percent Relocated	45%	44%	45%	35%/40%

We are confident that the staff's conclusions will provide an adequate basis for the Owners Groups to proceed with the development of complete new STS in accordance with the Commission's Interim Policy Statement.

APPENDIX A

RESULTS OF THE NRC STAFF REVIEW

BABCOCK & WILCOX OWNERS GROUP'S SUBMITTAL

RETENTION AND RELOCATION OF SPECIFIC TECHNICAL SPECIFICATIONS

APPENDIX A

TABLE 1

LCOs TO BE RETAINED IN BABCOCK & WILCOX STANDARD TECHNICAL SPECIFICATIONS

LCO		CRITERIA
3.1	REACTIVITY CONTROL SYSTEM	
3.1.1.1	Shutdown Margin (Note 1)	2
3.1.1.2	Moderator Temperature Coefficient	2
3.1.1.3	Minimum Temperature for Criticality	2
3.1.3.1	Group Height - Safety and Regulating Rod Groups	2
3.1.3.2	Group Height - Axial Power Shaping Rod Group	2
3.1.3.6	Safety Rod Insertion Limit	2 & 3
3.1.3.7	Regulating Rod Insertion Limits	2
3.1.3.9	Xenon Reactivity	2
3.2	POWER DISTRIBUTION LIMITS	
3.2.1	Axial Power Imbalance	2
3.2.2	Nuclear Heat Flux Hot Channel Factor	2
3.2.3	Nuclear Enthalpy Rise Hot Channel Factor	2
3.2.4	Quadrant Power Tilt	2
3.2.5	DNB Parameters	2
3.3	INSTRUMENTATION	
3.3.1	Reactor Protection System Instrumentation (Note 2)	3
3.3.2	Engineered Safety Feature Actuation System Instrumentation (Note 2)	3
3.3.3.1	Radiation Monitoring Instrumentation (Notes 2 & 3)	3
3.3.3.5	Remote Shutdown Instrumentation (Notes 2 & 4)	Risk
3.3.3.6	Accident Monitoring Instrumentation	3
3.4	REACTOR COOLANT SYSTEM	
3.4.1.1	Startup and Power Operation	3
3.4.1.2	Hot Standby	3
3.4.1.3	Hot Shutdown	3
3.4.1.4	Cold Shutdown	Policy Statement (DHR)
3.4.3	Safety Valve - Operating	3
3.4.4	Pressurizer	2 & 3
3.4.5	Relief Valve	3
3.4.6	Steam Generators - Water Level	2
3.4.7.1	Leakage Detection System	1

LCO		CRITERIA
3.4.7.2	Operational Leakage	2
..4.9	Specific Activity	2
3.4.10.1	Reactor Coolant System Pressure/Temperature Limits	2
3.4.10.3	Overpressure Protection System	2
3.5	EMERGENCY CORE COOLING SYSTEM (ECCS)	
3.5.1	Core Flooding Tanks	2 & 3
3.5.2	ECCS Subsystems - $T_{avg} \geq (305)°F$	3
3.5.3	ECCS Subsystems - $T_{avg} \leq (305)°F$	3
3.5.4	Borated Water Storage Tank	2 & 3
3.6	CONTAINMENT SYSTEMS	
3.6.1.1	Containment Integrity	3
3.6.1.3	Containment Air Locks	3
3.6.1.5	Internal Pressure	2
3.6.1.6	Air Temperature	2
3.6.1.8	Containment Ventilation System	3
3.6.2.1	Containment Spray System	3
3.6.2.2	Spray Additive System	2 & 3
3.6.2.3	Containment Cooling System	3
3.6.3	Iodine Cleanup System	3
3.6.4	Containment Isolation Valves	3
3.6.5.1	Hydrogen Analyzers	3
3.6.5.2	Electric Hydrogen Recombiners (Note 5)	3
3.6.6	Penetration Room Exhaust Air Cleanup System	3
3.7	PLANT SYSTEMS	
3.7.1.1	Safety Valves	3
3.7.1.2	Auxiliary Feedwater System	3
3.7.1.3	Condensate Storage Tank	2 & 3
3.7.1.4	Activity	2
3.7.1.5	Main Steam Line Isolation Valves	3
3.7.3	Component Cooling Water System	3
3.7.4	Service Water System	3
3.7.5	Ultimate Heat Sink	3
3.7.6	Flood Protection (optional)	3
3.7.7	Control Room Emergency Air Cleanup System	3
3.7.8	ECCS Pump Room Exhaust Air Cleanup System (optional)	3

B&W-TABLE 1 (Continued)

LCO		CRITERIA
3.8	ELECTRICAL POWER SYSTEMS	
3.8.1.1	A.C. Sources - Operating	3
3.8.1.2	A.C. Sources - Shutdown	Policy Statement (DHR)
3.8.2.1	A.C. Distribution - Operating	3
3.8.2.2	A.C. Distribution - Shutdown	Policy Statement (DHR)
3.8.2.3	D.C. Distribution - Operating	3
3.8.2.4	D.C. Distribution - Shutdown	Policy Statement (DHR)
3.9	REFUELING OPERATIONS	
3.9.1	Boron Concentration	2
3.9.2	Instrumentation	3
3.9.3	Decay Time	2
3.9.4	Containment Building Penetration	3
3.9.8.1	Residual Heat Removal and Coolant Circulation - All Water Levels	Policy Statement (DHR)
3.9.8.2	Residual Heat Removal and Coolant Circulation - Low Water Levels	Policy Statement (DHR)
3.9.9	Containment Purge and Exhaust Isolation System	3
3.9.10	Water Level - Reactor Vessel	2
3.9.11	Water Level - Storage Pool	2
3.9.12	Storage Pool Air Cleanup System	2

Notes:

1. Required for Modes 3 through 5. May be relocated for Modes 1 and 2.

2. The LCO for this system should be retained in STS. The Policy Statement criteria should not be used as the basis for relocating specific trip functions, channels, or instruments within these LCOs.

3. The staff is pursuing alternative approaches which would allow relocation of some of these LCOs on a schedule consistent with the schedule for development of the new STS. The staff is also initiating rulemaking to delete the requirement that RETS be included in Technical Specifications.

4. Because fires (either inside or outside the control room) can be a significant contributor to the core melt frequency and because the uncertainties with fire initiation frequency can be significant, the staff believes that this LCO should be retrained in the STS at this time. The staff will consider relocation of Remote Shutdown Instrumentation on a plant-specific basis.

5. This LCO will be considered for relocation to a licensee-controlled document on a plant-specific basis.

TABLE 2 (Note 1)

BABCOCK & WILCOX STANDARD TECHNICAL SPECIFICATION

LCOs WHICH MAY BE RELOCATED

LCO	
3.1	REACTIVITY CONTROL SYSTEMS
3.1.2.1	Flow Paths - Shutdown
3.1.2.2	Flow Paths - Operating
3.1.2.3	Makeup Pump - Shutdown
3.1.2.4	Makeup Pump - Operating
3.1.2.5	Decay Heat Removal Pump - Shutdown
3.1.2.6	Boric Acid Pumps - Shutdown
3.1.2.7	Boric Acid Pumps - Operating
3.1.2.8	Borated Water Source - Shutdown
3.1.2.9	Borated Water Source - Operating
3.1.3.3	Position Indication Channels - Operating (Note 2)
3.1.3.4	Position Indication Channels - Shutdown (Note 2)
3.1.3.5	Rod Drop Time (Note 2)
3.1.3.8	Rod Program
3.3	INSTRUMENTATION
3.3.3.2	Incore Detectors
3.3.3.3	Seismic Instrumentation
3.3.3.4	Meteorological Instrumentation
3.3.3.7	Chlorine Detection System
3.3.3.8	Fire Detection
3.3.3.9	Radioactive Liquid Effluent Monitor (Note 3)
3.3.3.10	Radioactive Gaseous Effluent Monitor (Note 3)
3.3.4	Turbine Overspeed Protection
3.4	REACTOR COOLANT SYSTEM
3.4.2	Safety Valves - Shutdown
3.4.6	Steam Generators Tube Surveillance (Note 4)
3.4.8	Chemistry
3.4.10.2	Pressurizer Temperatures
3.4.11	Structural Integrity ASME Code (Note 4)
3.4.12	RCS Vents
3.6	CONTAINMENT SYSTEMS
3.6.1.2	Containment Leakage (Note 5)
3.6.1.7	Containment Structural Integrity (Note 2)
3.7	PLANT SYSTEMS
3.7.2	Steam Generator Pressure/Temperature Limits
3.7.9	Snubbers
3.7.10	Sealed Source Contamination

B&W-TABLE 2 (Continued)

LCO	
3.7.11.1	Fire Suppression Water System
3.7.11.2	Spray and/or Sprinkler Systems
3.7.11.3	CO_2 System
3.7.11.4	Halon System
3.7.11.5	Fire Hose Stations
3.7.11.6	Yard Fire Hydrants and Hydrant Hose Houses
3.7.12	Fire Barrier Penetrations
3.7.13	Area Temperature Monitoring
3.9	REFUELING OPERATIONS
3.9.5	Communications
3.9.6	Fuel Handling Bridge
3.9.7	Crane Travel - Spent Fuel Storage Pool Building
3.10	SPECIAL TEST EXCEPTIONS
3.10.1	Shutdown Margin (Note 6)
3.10.2	Group Height Insertion Limits and Power Distribution Limits (Note 6)
3.10.3	Physics Tests (Note 6)
3.10.4	Reactor Coolant Loops (Note 6)
3.11	RADIOACTIVE EFFLUENTS (Note 3)
3.11.1.1	Concentration
3.11.1.2	Dose
3.11.1.3	Liquid Radwaste Treatment System
3.11.1.4	Liquid Holdup Tanks
3.11.2.1	Dose
3.11.2.2	Dose - Noble Gases
3.11.2.3	Dose - Iodine - 131, Tritium and Radionuclides in Particulate Form
3.11.2.4	Gaseous Radwaste Treatment Systems
3.11.2.5	Explosive Gas Mixture
3.11.2.6	Gas Storage Tanks
3.11.3	Solid Radioactive Waste
3.11.4	Total Dose
3.12	RADIOACTIVE ENVIRONMENTAL MONITORING (Note 3)
3.12.1	Monitoring Program
3.12.2	Land Use Census
3.12.3	Interlaboratory Comparison Program

B&W-TABLE 2 (Continued)

Notes:

1. Specifications listed in this table may be relocated contingent upon NRC staff approval of the location of and controls over relocated requirements.

2. This LCO may be removed from the STS. However, if the associated Surveillance Requirement(s) is necessary to meet the OPERABILITY requirements for a retained LCO, the Surveillance Requirement(s) should be relocated to the retained LCO.

3. The staff is pursuing alternative approaches which would allow relocation of some of these LCOs on a schedule consistent with the schedule for development of the new STS. The staff is also initiating rulemaking to delete the requirement that RETS be included in Technical Specifications.

4. This LCO may be relocated out of Technical Specifications. However, the associated Surveillance Requirement(s) must be relocated to Technical Specification Section 4.0, Surveillance Requirements.

5. This LCO may be relocated. However, Pa, La, Ld, and Lt must be either retained in TS or in the Bases of the appropriate Containment LCO.

6. Special Test Exceptions may be included with corresponding LCOs.

APPENDIX B

RESULTS OF THE NRC STAFF REVIEW

WESTINGHOUSE OWNERS GROUP'S SUBMITTAL

RETENTION AND RELOCATION OF SPECIFIC TECHNICAL SPECIFICATIONS

APPENDIX B

TABLE 1

LCOs TO BE RETAINED IN WESTINGHOUSE STANDARD TECHNICAL SPECIFICATIONS

LCO		CRITERIA
3.1	REACTIVITY CONTROL SYSTEMS	
3.1.1.1	Shutdown Margin - Tave $\geq$ 200 deg. F (Note 1)	2
3.1.1.2	Shutdown Margin - Tave $\leq$ 200 deg. F (Note 1)	2
3.1.1.3	Moderator Temperature Coefficient	2
3.1.1.4	Minimum Temperature for Criticality	2
3.1.3.1	Moveable Control Assemblies - Group Height	3
3.1.3.5	Shutdown Rod Insertion Limit	2
3.1.3.6	Control Rod Insertion Limits	2
3.2	POWER DISTRIBUTION LIMITS	
3.2.1	Axial Flux Difference	2
3.2.2	Heat Flux Hot Channel Factor	2
3.2.3	RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor	2
3.2.4	Quadrant Power Tilt Ratio	2
3.2.5	DNB Parameters	2
3.3.	INSTRUMENTATION	
3.3.1	Reactor Trip System Instrumentation (Note 2)	3
3.3.2	Engineered Safety Feature Actuation System Instrumentation (Note 2)	3
3.3.3.1	Radiation Monitoring Instrumentation (Notes 2 & 3)	1 & 3
3.3.3.5	Remote Shutdown Instrumentation (Notes 2 & 4)	Risk
3.3.3.6	Accident Monitoring Instrumentation	3
3.4	REACTOR COOLANT SYSTEM	
3.4.1.1	RCS Startup and Power Operation	3
3.4.1.2	RCS Hot Standby	3
3.4.1.3	RCS Hot Shutdown	3
3.4.1.4.1	RCS Cold Shutdown - Loops Filled	3
3.4.1.4.2	RCS Cold Shutdown - Loops Not Filled	3
3.4.1.5	RCS Isolated Loop (Optional)	2
3.4.1.6	RCS Isolated Loop Startup (Optional)	2
3.4.2.2	RCS Safety valves - Operation	3
3.4.3	Pressurizer	2 & 3
3.4.4	Relief Valves	3
3.4.6.1	Leakage Detection System	1
3.4.6.2	Operational Leakage	2
3.4.8	Specific Activity	2
3.4.9.1	Pressure/Temperature Limits - RCS	2
3.4.9.3	Overpressure Protection Systems	2

W-TABLE 1 (Continued)

LCO		CRITERIA
3.5	EMERGENCY CORE COOLING SYSTEMS	
3.5.1.1	Cold Leg Injection Accumulators	2 & 3
3.5.1.2	Upper Head Injection Accumulators (STS REV-5)	2 & 3
3.5.2	ECCS Subsystems, Tavg _ 350 deg F	3
3.5.3	ECCS Subsystems, Tavg _ 350 deg F	3
3.5.4.1	Boron Injection Tank	2 & 3
3.5.5	Refueling Water Storage Tank	2 & 3
3.6	CONTAINMENT SYSTEMS	
3.6.1.1	Containment Integrity	3
3.6.1.3	Containment Air Locks	3
3.6.1.4	Containment Isolation Valve and Channel Weld Pressurization System (Optional)	3
3.6.1.5	Internal Pressure	2
3.6.1.6	Air Temperature	2
3.6.1.8	Containment Ventilation System	3
3.6.1.9	Shield Building Air Cleanup System (Ice Condenser)	3
3.6.2.1	Containment Quench Spray System (Sub-ATM Containment)	3
3.6.2.1	Containment Spray System	3
3.6.2.2	Containment Recirculation Spray System (Sub-ATM Containment)	3
3.6.2.2	Spray Additive System (Optional)	2 & 3
3.6.2.3	Containment Cooling System (Optional)	3
3.6.3	Iodine Cleanup System (Optional)	3
3.6.4	Containment Isolation Valves (minus response time)	3
3.6.5.1	Hydrogen Monitors	3
3.6.5.2	Electric Hydrogen Recombiners (Note 5)	3
3.6.5.3	Hydrogen Control Distributed Ignition System (STS REV-5, Ice Condenser)	3
3.6.5.4	Hydrogen Mixing System (Optional)	3
3.6.6	Penetration Room Exhaust Air Cleanup System (Optional)	3
3.6.7	Vacuum Relief Valves	3
3.6.7.1	Ice Bed (Ice Condenser)	2 & 3
3.6.7.3	Ice Condenser Doors (Ice Condenser)	2 & 3
3.6.7.5	Divider Barrier Personnel Access Doors and Equipment Hatches (Ice Condenser)	2 & 3
3.6.7.6	Containment Air Recirculation Systems (Ice Condenser)	2 & 3
3.6.7.7	Floor Drains (Ice Condenser)	2 & 3
3.6.7.8	Refueling Canal Drains (Ice Condenser)	3
3.6.7.9	Divider Barrier Seal (Ice Condenser)	2 & 3
3.6.8.1	Shield Building Air Cleanup System (Dual)	3
3.6.8.2	Shield Building Integrity (Dual)	3

W-TABLE 1 (Continued)

LCO		CRITERIA
3.7	PLANT SYSTEMS	
3.7.1.1	Turbine Cycle Safety Valves	3
3.7.1.2	Auxiliary Feedwater System	2 & 3
3.7.1.3	Condensate Storage Tank	2 & 3
3.7.1.4	Activity	2
3.7.1.5	Main Steam Line Isolation Valves	3
3.7.3	Component Cooling Water System	3
3.7.4	Service Water System	3
3.7.5	Ultimate Heat Sink (Optional)	3
3.7.7	Control Room Emergency Air Cleanup System	3
3.7.8	ECCS Pump Room Emergency Air Cleanup System	3
3.8	ELECTRICAL POWER SYSTEMS	
3.8.1.1	A.C. Sources - Operating	3
3.8.1.2	A.C. Sources - Shutdown	3
3.8.2.1	D.C. Sources - Operating	3
3.8.2.2	D.C. Sources - Shutdown	3
3.8.3.1	Onsite Power Distribution - Operating	3
3.8.3.2	Onsite Power Distribution - Shutdown	3
3.9	REFUELING OPERATIONS	
3.9.1	Boron Concentration	2
3.9.2	Instrumentation	3
3.9.3	Decay Time	2
3.9.4	Containment Building Penetrations	3
3.9.8.1	Residual Heat Removal and Coolant Circulation - High Water Level	Policy Statement (RHR)
3.9.8.2	Residual Heat Removal and Coolant Circulation - Low Water Level	Policy Statement (RHR)
3.9.9	Containment Purge and Exhaust Isolation System	3
3.9.10	Water Level - Reactor Vessel	2
3.9.11	Water Level - Storage Pool	2
3.9.12	Storage Pool Air Cleanup System	3

Notes:

1. Required for Modes 3 through 5. May be relocated for Modes 1 and 2.

2. The LCO for this system should be retained in STS. The Policy Statement criteria should not be used as the basis for relocating specific trip functions, channels, or instruments within these LCOs.

3. The staff is pursuing alternative approaches which would allow relocation of some of these LCOs on a schedule consistent with the schedule for development of the new STS. The staff is also initiating rulemaking to delete the requirement that RETS be included in Technical Specifications.

W-TABLE 1 (Continued)

Notes:

4. Because fires (either inside or outside the control room) can be a significant contributor to the core melt frequency and because the uncertainties with fire initiation frequency can be significant, the staff believes that this LCO should be retained in the STS at this time. The staff will consider relocation of Remote Shutdown Instrumentation on a plant-specific basis.

5. This LCO will be considered for relocation to a licensee-controlled document on a plant-specific basis.

TABLE 2 (Note 1)

WESTINGHOUSE STANDARD TECHNICAL SPECIFICATIONS
LCOs WHICH MAY BE RELOCATED

LCO	
3.1	REACTIVITY CONTROL SYSTEMS
3.1.2.1	Flow Paths - Shutdown
3.1.2.2	Flow Paths - Operating
3.1.2.3	Charging Pumps - Shutdown
3.1.2.4	Charging pumps - Operating
3.1.2.5	Borated Water Sources - Shutdown
3.1.2.6	Borated Water Sources - Operating
3.1.3.2	Position Indication System - Operating (Note 2)
3.1.3.3	Position Indication System - Shutdown (Note 2)
3.1.3.4	Rod Drop Time (Note 2)
3.3	INSTRUMENTATION
3.3.3.2	Movable Incore Detectors
3.3.3.3	Seismic Instrumentation
3.3.3.4	Meteorological Instrumentation
3.3.3.7	Chlorine Detection Systems
3.3.3.8	Fire Detection Instrumentation
3.3.3.9	Loose-Part Detection Instrumentation
3.3.3.10	Radioactive Liquid Effluent Monitoring Instrumentation (Note 3)
3.3.3.11	Radioactive Gaseous Effluent Monitoring Instrumentation (STS REV - 5) (Note 3)
3.3.4	Turbine Overspeed Protection
3.4	REACTOR COOLANT SYSTEM
3.4.2.1	RCS Safety Valves - Shutdown
3.4.5	Steam Generators (Note 4)
3.4.7	Chemistry
3.4.9.2	Pressure/Temperature Limits - Pressurizer
3.4.10	RCS Structural Intgerity (Note 4)
3.4.11	Reactor Coolant System Vents (STS REV-5)
3.5	EMERGENCY CORE COOLING SYSTEMS
3.5.4.2	Heat Tracing

W-TABLE 2 (Continued)

LCO	
3.6	CONTAINMENT SYSTEMS
3.6.1.2	Containment Leakage (Note 5)
3.6.1.7	Containment Structural Integrity (Note 2)
3.6.1.8	Shield Building Structural Integrity (Ice Condenser) (Note 2)
3.6.4	Containment Isolation Valves (response times) (Note 2)
3.6.5.1	Steam Jet Air Ejector (Sub-ATM Containment)
3.6.5.2	Mechanical Vacuum Pumps (SUB-ATM. Containment)
3.6.5.3	Hydroden Purge Cleanup System
3.6.7.2	Ice Bed Temperature Monitoring System (Ice Condenser)
3.6.7.4	Inlet Door Position Monitoring System (Ice Condenser)
3.6.8.3	Shield Building Structural Integrity (Dual)
3.7	PLANT SYSTEMS
3.7.2	Steam Generator Pressure/Temperature Limitation
3.7.6	Flood Protection (Optional)
3.7.9	Snubbers
3.7.10	Sealed Source Contamination
3.7.11.1	Fire Suppression Water System
3.7.11.2	Spray and/or Sprinkler Systems
3.7.11.3	CO2 Systems
3.7.11.4	Halon Systems
3.7.11.5	Fire Hose Stations
3.7.11.6	Yard Fire Hydrants and Hydrant Hose Houses
3.7.12	Fire Rated Assemblies
3.7.13	Area Temperature Monitoring
3.8	ELECTRICAL POWER SYSTEMS
3.8.4.1	A.C. Circuits Inside Primary Containment (STS REV-5)
3.8.4.2	Containment Penetration Conductor Overcurrent Protective Devices
3.8.4.3	Motor-Operated Valves Thermal Overload Protection and Bypass Devices
3.9	REFUELING OPERATIONS
3.9.5	Communications
3.9.6	Manipulator Crane
3.9.7	Crane Travel - Spent Fuel Storage Pool
3.10	SPECIAL TEST EXCEPTIONS (Note 6)

W-TABLE 2 (Continued)

LCO	
3.11	RADIOACTIVE EFFLUENTS (Note 3)
3.11.1.1	Liquid Effluents Concentration (STS REV-5)
3.11.1.2	Dose (STS REV-5)
3.11.1.3	Liquid Radwaste Treatment System (STS REV-5)
3.11.1.4	Liquid Holdup Tanks (STS REV-5)
3.11.2.1	Dose Rate (STS REV-5)
3.11.2.2	Dose - Noble Gases (STS REV-5)
3.11.2.3	Dose I-131, I-133, Tritium and Radioactive Material In Particulate Form
3.11.2.4	Gaseous Radwaste Treatment (STS REV-5)
3.11.2.5	Explosive Gas Mixture (STS REV-5)
3.11.2.6	Gas Storage Tanks
3.11.3	Solid Radioactive Waste (STS REV-5)
3.11.4	Total Dose (STS REV-5)
3.12	RADIOLOGICAL ENVIRONMENTAL MONITORING (Note 3)
3.12.1	Monitoring Program (STS REV-5)
3.12.2	Land Use Census (STS REV-5)
3.12.3	Interlaboratory Comparison Program (STS REV-5)

Notes:

1. LCOs listed in this table may be relocated contingent upon NRC staff approval of the location of and controls over relocated requirements.

2. This LCO may be removed from the STS. However, if the associated Surveillance Requirement(s) is necessary to meet the OPERABILITY requirements for a retained LCO, the Surveillance Requirement(s) should be relocated to the retained LCO.

3. The staff is pursuing alternative approaches which would allow relocation of some of these LCOs on a schedule consistent with the schedule for development of the new STS. The staff is also initiating rulemaking to delete the requirement that RETS be included in Technical Specifications.

4. This LCO may be relocated out of Technical Specifications. However, the associated Surveillance Requirement(s) must be relocated to Technical Specification Section 4.0, Surveillance Requirements.

5. This LCO may be relocated. However, Pa, La, Ld and Lt must be either retained in TS or in the Bases of the appropriate containment LCO.

6. Special Test exceptions 3.10.1 through 3.10.4 may be included with corresponding LCOs which are remaining in Technical Specifications. Special Test Exception 3.10.5 may be relocated outside of Technical Specifications along with LCO 3.1.3.3.

APPENDIX C

RESULTS OF THE NRC STAFF REVIEW

COMBUSTION ENGINEERING OWNERS GROUP'S SUBMITTAL

RETENTION AND RELOCATION OF SPECIFIC TECHNICAL SPECIFICATIONS

APPENDIX C

TABLE 1

LCOs TO BE RETAINED IN COMBUSTION ENGINEERING STANDARD TECHNICAL SPECIFICATIONS

LCO		CRITERIA
3.1	REACTIVITY CONTROL SYSTEMS	
3.1.1.1	Shutdown Margin --Tcold. $\geq$ 210F (Note 1)	2
3.1.1.2	Shutdown Margin - Tcold. $\leq$ 210F (Note 1)	2
3.1.1.3	Moderator Temperature Coefficient	2
3.1.1.4	Minimum Temperature for Criticality	2
3.1.3.1	CEA Position	2 & 3
3.1.3.5	Shutdown CEA Insertion Limit	2
3.1.3.6	Regulating CEA Insertion Limits	2
3.1.3.7	Part Length CEA Insertion Limits	2
3.2	POWER DISTRIBUTION LIMITS	
3.2.1	Linear Heat Rate	2
3.2.2	Planar Radial Peaking Factors--Fxy	2
3.2.3	Azimuthal Power Tilt -- Tq	2
3.2.4	DNBR Margin	2
3.2.5	RCS Flow Rate	2
3.2.6	Reactor Coolant Cold Leg Temperature	2
3.2.7	Axial Shape Index	2
3.2.8	Pressurizer Pressure	2
3.3	INSTRUMENTATION	
3.3.1	Reactor Protective Instrumentation (Note 2)	3
3.3.2	ESFAS Instrumentation (Note 2)	3
3.3.3.1	Radiation Monitoring Instrumentation (Notes 2 & 3)	3
3.3.3.5	Remote Shutdown System (Notes 2 & 4)	Risk
3.3.3.6	Post-Accident Monitoring Instrumentation	3
3.4	REACTOR COOLANT SYSTEM	
3.4.1.1	Startup and Power Operation	2 & 3
3.4.1.2	Hot Standby	2 & 3
3.4.1.3	Hot Shutdown	2 & 3
3.4.1.4.1	Cold Shutdown - Loops filled	2 & 3
3.4.1.4.2.	Cold Shutdown - Loops not filled	2 & 3

CE-TABLE 1 (Continued)

LCO		CRITERIA
3.4.2.2	Safety Valves - Operating	3
3.4.3.1	Pressurizer	2 & 3
3.4.4	Relief Valve (PORV Only)	3
3.4.6.1	Leakage Detection Systems	3
3.4.6.2	Operational Leakage	3
3.4.8	Specific Activity	2
3.4.9.1	Reactor Coolant System	2
3.4.9.3	Overpressure Protection Systems-LTOP	2
3.5	EMERGENCY CORE COOLING SYSTEMS (ECCS)	
3.5.1	Safety Injection Tanks	3
3.5.2	ECCS Subsystems -- Tcold. ≥ 350F	3
3.5.3	ECCS Subsystems -- Tcold. ≤ 350F	3
3.5.4	Refueling Water Tank	3
3.6	CONTAINMENT SYSTEMS	
3.6.1.1	Containment Integrity	3
3.6.1.3	Containment Air Locks	3
3.6.1.5	Internal Pressure	2
3.6.1.6	Air Temperature	2
3.6.1.8	Containment Ventilation System (Optional)	3
3.6.2.1	Containment Spray System	3
3.6.2.2	Spray Additive System (Optional)	3
3.6.2.3	Containment Cooling System (Optional)	3
3.6.3	Iodine Cleanup System (Optional)	3
3.6.4	Containment Isolation Valves	3
3.6.5.1	Hydrogen Monitors (Note 5)	3
3.6.5.2	Electric Hydrogen Combiners (Note 5)	3
3.6.5.4	Hydrogen Mixing System	3
3.6.6	Penetration Room Exhaust Air Cleanup System (Optional)	3
3.6.7	Vacuum Relief Valves (Optional)	3
3.6.8.1	Shield Building Air Cleanup System (Optional)	3
3.7	PLANT SYSTEMS	
3.7.1.1	Safety Valves	3
3.7.1.2	Auxiliary Feedwater System	3
3.7.1.3	Condensate Storage Tank	3
3.7.1.4	Activity	3
3.7.1.5	Main Steam Isolation Valves	3

CE-TABLE 1 (Continued)

LCO		CRITERIA
3.7.3	Component Cooling Water System	3
3.7.4	Service Water System	3
3.7.5	Ultimate Heat Sink	3
3.7.7	Essential Chilled Water System	3
3.7.9	ECCS Pump Room Air Exhaust Cleanup System (Optional)	3
3.8	ELECTRICAL POWER SYSTEMS	
3.8.1.1	A.C. Sources - Operating	3
3.8.1.2	A.C. Sources - Shutdown	3
3.8.2.1	D.C. Sources - Operating	3
3.8.2.2	D.C. Sources - Shutdown	3
3.8.3.1	Onsite Power Distribution Sources - Operating	3
3.8.3.2	Onsite Power Distribution Sources - Shutdown	3
3.9	REFUELING OPERATIONS	
3.9.1	Boron Concentration	2
3.9.2	Instrumentation	3
3.9.3	Decay Time	2
3.9.4	Containment Building Penetrations	3
3.9.8.1	Shutdown Cooling and Coolant Circulation - High Water Level	2
3.9.8.2	Shutdown Cooling and Coolant Circulation - Low Water Level	2
3.9.9	Containment Purge Valve Isolation System	3
3.9.10	Water Level-Reactor Vessel	2
3.9.11	Water Level-Storage Pool	2
3.9.12	Fuel Building Air Cleanup System	3

Notes:

1. Required for Modes 3 through 5. May be relocated for Modes 1 and 2.

2. LCOs for this system should be retained in STS. The Policy Statement Criteria should not be used to relocate specific trip functions, channels, or instruments within these LCOs.

3. The staff is pursuing alternative approaches which would allow relocation of some of these LCOs on a schedule consistent with the schedule for development of the new STS. The staff is also initiating rulemaking to delete the requirement that RETS be included in Technical Specifications.

4. Because fires (either inside or outside the control room) can be a significant contributor to the core melt frequency and because the uncertainties with fire initiation frequency can be significant, the staff believes that this LCO should be retained in the STS at this time. The staff will consider relocation of Remote Shutdown Instrumentation on a plant-specific basis.

5. This LCO will be considered for relocation to a licensee-controlled document on a plant-specific basis.

TABLE 2 (Note 1)

COMBUSTION ENGINEERING STANDARD TECHNICAL SPECIFICATION
LCOs WHICH MAY BE RELOCATED

LCO	
3.1	REACTIVITY CONTROL SYSTEMS
3.1.2.1	Flow Paths -- Shutdown
3.1.2.2	Flow Paths-Operating
3.1.2.3	Charging Pumps -- Shutdown
3.1.2.4	Charging Pumps-Operating
3.1.2.5	Boric Acid Makeup Pumps -- Shutdown
3.1.2.6	Boric Acid Makeup Pumps-Operating
3.1.2.7	Borated Water Source - Shutdown
3.1.2.8	Borated Water Sources - Operating
3.1.3.2	Position Indicator Channels-Operating (Note 2)
3.1.3.3	Position Indicator Channels-Shutdown (Note 2)
3.1.3.4	CEA Drop Time (Note 2)
3.3	INSTRUMENTATION
3.3.3.2	Incore Detectors
3.3.3.3	Seismic Instrumentation
3.3.3.4	Meteorological Instrumentation
3.3.3.7	Fire Detection Instrumentation
3.3.3.8	Chlorine Detection Systems
3.3.3.9	Loose Part Detection Instrumentation
3.3.3.10	Radioactive Liquid Effluent Monitor (Note 3)
3.3.3.11	Radioactive Gaseous Effuent Monitor (Note 3)
3.3.4	Turbine Overspeed Protection
3.4	REACTOR COOLANT SYSTEM
3.4.2.1	Safety Valves-Shutdown
3.4.4	Relief Valves (Non PORV)
3.4.5	Steam Generators (Note 4)
3.4.7	Chemistry
3.4.9.2	Pressurizer Heatup/Cooldown Limits
3.4.10	Structural Integrity (Note 4)
3.4.11	Reactor Coolant System Vents
3.6	CONTAINMENT SYSTEMS
3.6.1.2	Containment Leakage (Note 5)
3.6.1.4	Containment Isolation Valve and Channel Weld Pressure System
3.6.1.7	Containment Vessel Structural Integrity (Note 2)
3.6.5.3	Hydrogen Purge Cleanup System
3.6.8.2	Shield Building Integrity
3.6.8.3	Shield Building Structural Integrity (Note 2)

CE-TABLE 2 (Continued)

LCO	
3.7	PLANT SYSTEMS
3.7.2	Steam Generator Pressure/Temperature Limitation
3.7.6	Flood Protection
3.7.8	Control Room Emergency Air Cleanup System
3.7.10	Snubbers
3.7.11	Sealed Source Contamination
3.7.12	Fire Suppression Systems
3.7.12.1	Fire Suppression Water System
3.7.12.2	Spray and/or Sprinkler Systems
3.7.12.3	CO2 Systems
3.7.12.4	Halon Systems
3.7.12.5	Fire Hose Stations
3.7.12.6	Yard Fire Hydrants and Hose Houses
3.7.13	Fire-Rated Assemblies
3.8	ELECTRICAL POWER SYSTEMS
3.8.4.1	Containment Penetration Conductor Overcurrent Protection Device
3.8.4.2	Motor-Operated Valves-Thermal Overload Protection
3.9	REFUELING OPERATIONS
3.9.5	Communication
3.9.6	Manipulator Crane (Refueling Machine)
3.9.7	Crane Travel - Spent Fuel Pool Building
3.10	SPECIAL TEST EXCEPTIONS
3.10.1	Shutdown Margin (Note 6)
3.10.2	Group Height, Insertion, and Power Dist. (Note 6)
3.10.3	Reactor Coolant Loops (Note 6)
3.10.4	CEA Position, Reg CEA Ins, and Cold Leg Temp. (Note 6)
3.11	RADIOACTIVE EFFLUENTS (Note 3)
3.11.1.1	Liquid Waste Discharge to Evap. Ponds - Concentration
3.11.1.2	Liquid Waste Discharge to Evap. Ponds Dose
3.11.1.3	Liquid Holdup Tanks
3.11.2.1	Gaseous Effluents - Dose Rate
3.11.2.2	Gaseous Effluents - Dose-Noble Gases
3.11.2.3	Gaseous Effluents - Dose--I-131, 133, Tritium & Radionuclides
3.11.2.4	Gaseous Radwaste Treatment
3.11.2.5	Explosive Gas Mixture
3.11.2.6	Gas Storage Tanks
3.11.3	Solid Radioactive Waste
3.11.4	Total Dose

CE-TABLE 2 (Continued)

LCO	
3.12	RADIOLOGICAL ENVIRONMENTAL MONITORING (Note 3)
3.12.1	Monitoring Program
3.12.2	Land Use Census
3.12.3	Interlaboratory Comparison Program

Notes:

1. Specifications listed in this table may be relocated contingent upon NRC staff approval of the location of and controls over relocated requirements.

2. This LCO may be removed from the STS. However, if the associated Surveillance Requirement(s) is necessary to meet the OPERABILITY requirements for a retained LCO, the Surveillance Requirement(s) should be relocated to the retained LCO.

3. The staff is pursuing alternative approaches which would allow relocation of some of these LCOs on a schedule consistent with the schedule for development of the new STS. The staff is also initiating rulemaking to delete the requirement that RETS be included in Technical Specifications.

4. This LCO may be relocated out of Technical Specifications. However, the associated Surveillance Requirement(s) must be relocated to Technical Specification Section 4.0, Surveillance Requirements.

5. This LCO may be relocated. However, Pa, La, Ld, and Lt must be either retained in TS or in the Bases of the appropriate containment LCO.

6. Special Test Exceptions may be included with the corresponding LCOs.

APPENDIX D

RESULTS OF THE NRC STAFF REVIEW

BWR OWNERS GROUP'S SUBMITTAL

RETENTION AND RELOCATION OF SPECIFIC TECHNICAL SPECIFICATIONS

APPENDIX D

TABLE 1

LCOs TO BE RETAINED IN GENERAL ELECTRIC STANDARD TECHNICAL SPECIFICATIONS

LCO	REPORT ITEM		PLANT*	CRITERIA
3.1		REACTIVITY CONTROL SYSTEMS		
3.1.1	1	Shutdown Margin	H,GG	2
3.1.3		Control Rods		
	3	Control Rods Operability	H,GG	3
	5	Maximum Scram Times (BWR/6)	GG	3
	6	Average Scram Times	H	3
	7	Fastest 3-out-of-4 Scram Times	H	3
	8	Scram Accumulators	H,GG	3
	9	Control Rod Drive Coupling	H,GG	3
	10	Control Rod Position Indication	H,GG	3
	11	Control Rod Drive Housing Support	H,GG	3
3.1.4		Control Rod Program Controls		
	12	Rod Worth Minimizer (BWR/2-5)	H	3
	13	Control Rod Withdrawal (BWR/6)	GG	2
	14	Rod Pattern Control System (BWR/6)	GG	3
	15	Rod Sequence Control Systems	H	3
	16	Rod Block Monitor	H	3
3.1.5	17	Standby Liquid Control System	H,GG	Policy Statement(SBLC)
3.1.6	18	Scram Discharge Volume Vent and Drain Valves	H	3
3.2		POWER DISTRIBUTION LIMITS		
3.2.1	19	Average Planar Linear Heat Generation (APLHGR)	H,GG	2
3.2.3	21	Minimum Critical Power Ratio (MCPR)	H,GG	2
3.2.4	22	Linear Heat Generation Rate (LHGR)	H,GG	2

*H-Hatch Unit 2
GG-Grand Gulf

BWR-TABLE 1 (Continued)

LCO	REPORT ITEM		PLANT	CRITERIA
3.3		INSTRUMENTATION		
3.3.1		Reactor Protection System Instrumentation (Note 1)		
	23	Average Power Range Monitors (APRM)	H,GG	3
	24	Intermediate Range Monitors (IRM)	H,GG	3
	25	Vessel Pressure - High	H,GG	3
	26	Reactor Vessel Water Level - Low (Level 3)	H,GG	3
	27	Reactor Vessel Water Level - High (Level 8)	GG	3
	28	MSIV Closure	H,GG	3
	29	MSL Radiation - High (RPS Instr)	H,GG	3
	30	Drywell Pressure - High	H,GG	3
	31	SDV Water Level - High	H,GG	3
	32	TSV Closure	H,GG	3
	33	TCV Closure	H,GG	3
	34	Mode Switch	H,GG	3
	35	Manual Scram	H,GG	3
3.3.2		Isolation Actuation Instrumentation (Note 1)		
		Primary Containment Isolation		
	36	Reactor Vessel Water Level - Low (Level 3)	H	3
	37	Reactor Vessel Water Level - Low (Level 2)	H,GG	3
	38	Reactor Vessel Water Level - Low (Level 1)	H,GG	3
	39	Drywell Pressure - High	H,GG	3
	40	Containment and Drywell Ventilation Exhaust Radiation - High High	GG	3
		Main Steam Line Isolation		
	41	Manual Initiation (Primary Containment)	GG	3
	42	Reactor Vessel Water Level - Low (Level 1)	GG	3
	43	Main Steam Line Radiation - High (MSLI)	H,GG	3
	44	Main Steam Line Pressure - Low	H,GG	3
	45	Main Steam Line Flow - High	H,GG	1 & 3

BWR-TABLE 1 (Continued)

LCO	REPORT ITEM		PLANT	CRITERIA
	46	Condenser Vacuum - Low	H,GG	3
	47	Main Steam Line Tunnel Temperature - High	H,GG	1 & 3
	48	Main Steam Line Tunnel Differential Temperature - High	GG	1 & 3
	49	Manual Initiation (MSLI)	GG	3
	50	Turbine Building Area Temperature - High	H	1 & 3
		Secondary Containment Isolation		
	51	Reactor Building Exhaust Radiation - High	H	3
	52	Reactor Vessel Water Level - Low (Level 2)	H,GG	3
	53	Drywell Pressure - High	H,GG	3
	54	Refueling Floor Exhaust Radiation - High	H	3
	55	Fuel Handling Area Ventilation Exhaust Radiation - High High	GG	3
	56	Fuel Handling Area Pool Sweep Exhaust Radiation - High High	GG	3
		Reactor Water Cleanup System Isolation		
	57	Manual Initiation (Secondary Containment)	GG	3
	58	Differential Flow - High	H,GG	1 & 3
	59	Differential Flow Timer	GG	2
	60	Equipment Area Temperature - High	H,GG	1 & 3
	61	Equipment Area Differential Temperature - High	H,GG	1 & 3
	62	Reactor Vessel Water Level - (Level 2)	H,GG	3
	63	Main Steam Line Tunnel Temperature - High	GG	1 & 3
	64	Main Steam Line Tunnel Differential Temperature - High	GG	1 & 3
	65	SLCS Initiation	H,GG	Policy Statement (SBLC

BWR-TABLE 1 (Continued)

LCO	REPORT ITEM		PLANT	CRITERIA
		High Pressure Coolant Injection System Isolation		
	66	Manual Initiation (RWCS)	GG	3
	67	HPCI Steam Line Flow - High	H	1 & 3
	68	HPCI Steam Supply Pressure - Low	H	3
	69	HPCI Turbine Exhaust Diaphragm Pressure - High	H	3
	70	HPCI Pipe Penetration Room Temperature - High	H	1 & 3
	71	Suppression Pool Area Ambient Temperature - High	H	1 & 3
	72	Suppression Pool Area Differential Temperature - High	H	1 & 3
	73	Suppression Pool Area Temperature Timer Relays	H	2 & 3
	74	Emergency Area Cooler Temperature - High	H	1 & 3
	76	Logic Power Monitor	H	3
		Reactor Core Isolation Cooling System Isolation		
	77	RCIC Steam Line Flow - High	H,GG	1 & 3
	78	RCIC Steam Supply Pressure - Low	H,GG	Policy Statement (RCIC)
	79	RCIC Turbine Exhaust Diaphragm Pressure - High	H,GG	Policy Statement (RCIC)
	80	RCIC Equipment Area Temperature - High	H,GG	1 & 3
	81	Suppression Pool Area Ambient Temperature - High	H	1 & 3
	82	Suppression Pool Area Differential Temperature - High	H	1 & 3
	83	Suppression Pool Area Temperature Timer Relays	H	2 & 3
	85	Logic Power Monitor	H	3
	86	RCIC Equipment Room Differential Temperature - High	GG	1 & 3
	87	Main Steam Line Tunnel Temperature - High	GG	1 & 3
	88	Main Steam Line Tunnel Differential Temperature - High	GG	1 & 3

BWR-TABLE 1 (Continued)

LCO	REPORT ITEM		PLANT	CRITERIA
	89	Main Steam Line Tunnel Temperature Timer	GG	3
	90	RHR Equipment Room Temperature - High	GG	1 & 3
	91	RHR Equipment Room Differential Temperature - High	GG	1 & 3
	92	RHR/RCIC Steam Line Flow - High	GG	1 & 3
		RHR System Isolation		
	93	Manual Initiation (RCIC)	GG	3
	94	RHR Equipment Area Temperature - High	GG	1 & 3
	95	RHR Equipment Room Differential Temperature - High	GG	1 & 3
	96	Reactor Vessel Water Level - Low (Level 3)	H,GG	3
	97	Reactor Vessel (RHR Cut-In Permissive) Pressure - High	H,GG	Policy Statement (RHR)
	98	Drywell Pressure - High	GG	Policy Statement (RHR)
	99	Manual Initiation (RHR)	GG	
3.3.3		ECCS Actuation Instrumentation (Note 1) RHR (LPCI/LPCS/Core Spray)		
	100	Reactor Vessel Water Level - Low (Level 1)	H,GG	3
	101	Drywell Pressure - High	H,GG	3
	102	RHR Pump Time Delay	H,GG	3
	103	Manual Initiation	GG	3
		RHR (LPCI/LPCS/Core Spray)		
	104	Reactor Steam Dome Pressure - Low	H,GG	3
	105	Reactor Vessel Shroud Level - Low	H	3
	106	Logic Power Monitor	H	3
		Automatic Depressurization System		
	106A	Control Power Monitor	H	3
	107	Reactor Vessel Water Level Low (Level 1)	H,GG	3
	108	Drywell Pressure - High	H,GG	3
	109	ADS Initiation Timer	H,GG	3
	110	Low Water Level Timer	H	3

BWR-TABLE 1 (Continued)

LCO	REPORT ITEM		PLANT	CRITERIA
	111	Reactor Vessel Water Level Low (Level 3)	H,GG	3
	112	LPCI/LPCS/Core Spray Discharge Pressure - High	H,GG	3
	112A	ADS Bypass Timer	GG	3
		High Pressure Core Spray		
	112B	Manual Inhibit (ADS)	GG	3
	113	Manual Initiation (ADS)	GG	3
	114	Drywell Pressure - High	GG	3
	115	Reactor Vessel Water Level Low (Level 2)	GG	3
	116	Reactor Vessel Water Level High (Level 8)	GG	2
	117	CST Level - Low	GG	3
	118	Supp. Pool Water Level - High	GG	3
		HPCI		
	119	Manual Initiation (HPCS)	GG	3
	120	Drywell Pressure - High	H	3
	121	Reactor Vessel Water Level - Low (Level 2)	H	3
	122	Reactor Vessel Water Level - High (Level 8)	H	2
	123	Condensate Storage Tank Level - Low	H	3
	124	Suppression Chamber Water Level - High	H	3
	106	Logic Power Monitor	H	3
		ECCS Inst.		
	125	Loss of Power	GG	3
	126	Reactor Pressure - High (Low Low Set Interlock)	H	3
3.3.4		Recirculation Pump Trip Actuation Instrumentation		
	127	EOC-RPT	H,GG	3
	128	ATWS-RPT	H,GG	Policy Statement (RPT)
3.3.5		RCIC Instrumentation		
	129	Reactor Vessel Water Level - Low (Level 2)	H,GG	Policy Statement (RCI(
	130	Reactor Vessel Water Level - High (Level 8)	GG	Policy Statement (RCI(

BWR-TABLE 1 (Continued)

LCO	REPORT ITEM		PLANT	CRITERIA
	131	CST Level - Low	H,GG	Policy Statement (RCIC)
	132	Supp. Pool Water Level - High	H,GG	3
	133	Manual Initiation (RCIC)	GG	2
3.3.6		Control Rod Withdrawal Block Instrumentation		
	134	Rod Pattern Control System	GG	3
	136	RBM	H	3
	141	Reactor Mode Switch Shutdown Position	GG	3
3.3.7		Monitoring Instrumentation		
	142-150	Radiation Monitoring Instrumentation (Notes 1 & 2)		
	153	Remote Shutdown Instrumentation (Notes 1 & 3)	H,GG	Risk
	154-181	Accident Monitoring Instrumentation	H,GG	1, 2 & 3
	182	SRM	H,GG	2
3.3.8		Plant Systems Actuation Instrumentation		
	190	Drywell Press (Cont. Spray)	GG	3
	191	Cont. Press (Cont. Spray)	GG	3
	192	Water Level 1 (Cont. Spray)	GG	3
	193	Timers (Cont. Spray)	GG	3
	194	Water Level 8 (FW/TT)	GG	2
	195	Drywell Pressure (Supp. Pool Makeup System-SPMS)	GG	3
	196	Level 1 (SPMS)	GG	3
	197	Level 2 (SPMS)	GG	3
	198	Supp. Pool Level (SPMS)	GG	3
	199	Supp. Pool Makeup Timer (SPMS)	GG	3
	200	Manual Initiation (SPMS)	GG	3
3.3.10	201A	Neutron Flux Monitoring	GG	2
3.3.11	202	Degraded Voltage	H	3
3.4		REACTOR COOLANT SYSTEM		
3.4.1	203	Recirculation Loops	H,GG	2
	204	Jet Pumps	H,GG	3
	205	Idle Recirculation Loop Startup	H,GG	2
	206	Recirculation Loop Flow	GG	2

BWR-TABLE 1 (Continued)

LCO	REPORT ITEM		PLANT	CRITERIA
3.4.2	207	Safety/Relief Valves	H,GG	3
	208	S/RV Low-Low Set	H,GG	3
3.4.3	209	Leak Detection Systems	H,GG	1
3.4.3	210	Operational Leakage Limits	H,GG	1
3.4.5	212	Specific Activity	H,GG	2
3.4.6	213	Pressure/Temperature Limits		
	214	Reactor Steam Dome Pressure	H,GG	2
3.4.7	215	MSIVs	H,GG	3
3.4.9	217	RHR - Hot Shutdown	GG	Policy Statement (RHR
	218	RHR - Cold Shutdown	GG	Policy Statement (RHR
3.5		EMERGENCY CORE COOLING SYSTEMS		
3.5.1	219	HPCI	H	3
3.5.2	220	ADS	H	3
3.5.3	221	CSS	H	3
	222	LPCI	H	3
3.5.4	223	Supp. Pool	H,GG	3
	224	ECCS - Operating	GG	3
	225	ECCS - Shutdown	GG	3
3.6		CONTAINMENT SYSTEMS		
3.6.1		Primary Containment		
	226	Cont. Integrity	H,GG	3
	228	Air Locks	H,GG	3
	229	MSLIV-LCS	H,GG	3
	231	Structural Integrity	H,GG	3
	232	Cont. Internal Pressure	H,GG	2
	233	Cont. Air Temp	GG	2
	234	Containment Purge System	H,GG	3
3.6.2		Drywell		
	235	Drywell Integrity	H,GG	3
	236	Drywell Air Temperature	H,GG	2
	237	Drywell Bypass Leakage	GG	2
	238	Drywell Air Locks	GG	3
	239	Drywell Structural Integrity	GG	3
	240	Drywell Internal Pressure	GG	2
	241	Drywell Vent and Purge	GG	2

BWR-TABLE 1 (Continued)

LCO	REPORT ITEM		PLANT	CRITERIA
3.6.3		Depressurization Systems		
	242	Cont. Spray	GG	3
	243	Suppression Chamber (Pool)	H,GG	2 & 3
	244	Suppression Pool Makeup	GG	3
	245	Suppression Pool Cooling	H,GG	3
3.6.4	246	Isolation Valves	H,GG	3
3.6.5	247	Supp. Chamber - Drywell VB	H	3
	248	RB - Supp. Chamber VB	H	3
	249	Drywell Post LOCA VB	GG	3
3.6.6		Secondary Containment		
	250	Secondary Containment Integrity	H,GG	3
	251	Auto Isolation Dampers	H,GG	3
3.6.7		Containment Atmosphere Control		
	252	SGTS	H,GG	3
	253	H_2 Recombiner (Note 4)	H,GG	3
	254	H_2 Mixing System	H	3
	255	O_2 Conc.	H	3
	256	H_2 Ignition System	GG	3
3.7		PLANT SYSTEMS		
3.7.1	258	RHR Service Water	H	3
	259	Standby Service Water	GG	3
	260	Plant Service Water	H	3
	261	HPCS Service Water	GG	3
	262	Ultimate Heat Sink	GG	3
3.7.2	263	Control Room Environmental Control	H	3
	264	Control Room Emergency Filter	GG	3
3.7.3	265	RCIC	H,GG	Policy Statement (RCI(
3.8		ELECTRICAL POWER SYSTEMS		
3.8.1	274	Electrical Power Systems (AC/DC Sources, On-Site Distribution) (6 Sections)	H,GG	3
3.8.4	277	Power Monitoring of RPS	H,GG	3
	278	MOV Thermal Overload Protection	GG	3

BWR-TABLE 1 (Continued)

LCO	REPORT ITEM		PLANT	CRITERIA
3.9		REFUELING OPERATIONS		
3.9.1	279	Mode Switch	H,GG	3
	280	Instrumentation	H,GG	2
3.9.3	281	Control Rod Position	H,GG	2
3.9.4	282	Decay Time	H,GG	2
3.9.5	283	Secondary Cont. - Refueling Floor	H	3
	284	Secondary Cont. Isolation Dampers	H	3
	285	Standby Gas Treatment System	H	3
3.9.8	288	Crane Travel Spent Fuel Pool	H,GG	2
3.9.9	289	Water Level Reactor Vessel	H,GG	2
	290	Water Level Spent Fuel Pool	H,GG	2
	292	Coolant Circulation - High Water Level	H,GG	Policy Statement (RHR)
	293	Low Water Level	GG	Policy Statement (RHR)
3.11		RADIOACTIVE EFFLUENTS		
3.11.2	307	Main Condenser	H,GG	2

Notes:

1. LCOs for these systems should be retained in STS. The Policy Statement criteria should not be used to relocate specific trip functions, channels or instrument within these LCOs.

2. The staff is pursuing alternative approaches which would allow relocation of some of these LCOs on a schedule consistent with the schedule for development of the new STS. The staff is also initiating rulemaking to delete the requirement that RETS be included in Technical Specifications.

3. Because fires (either inside or outside the control room) can be a significant contributor to the core melt frequency and because the uncertainties with fire initiation frequency can be significant, the staff believes that this LCO should be retained in the STS at this time. The staff will consider relocation of Remote Shutdown Instrumentaiton on a plant-specific basis.

4. This LCO will be considered for relocation to a licensee-controlled document on a plant-specific basis.

BWR-TABLE 2 (Note 1)

GENERAL ELECTRIC STANDARD TECHNICAL SPECIFICATION
LCOs WHICH MAY BE RELOCATED

LCO	REPORT ITEM		PLANT
3.1		REACTIVITY CONTROL SYSTEMS	
3.1.2	2	Reactivity Anomaly (Note 2)	H,GG
3.1.3	4	Maximum Scram Times (7 Sec)	H
3.3		INSTRUMENTATION	
3.3.2		Isolation Actuation Instrumentation	
	75	Drywell Pressure - High (HPCI)	H
	84	Drywell Pressure - High (RCIC)	H,GG
3.3.6		Control Rod Withdrawal Block Instrumentation	
	135	APRM	H,GG
	137	SRM	H
	138	IRM	H,GG
	139	SDV Water Level	H,GG
	140	Reactor Coolant System Recirculation Flow-Upscale	GG
3.3.7		Monitoring Instrumentation	
	151	Seismic Monitors	H,GG
	152	Meteorological Inst.	GG
	183	TIP	H,GG
	184	Main Control Room Environmental System (Chlorine and Ammonia) Detection System	H
	186	Fire Protection	GG
	187	Loose-Parts	GG
	188	Radioactive Liquid Effluent (Note 3) Monitoring Instrumentation	H,GG
	189	Radioactive Gaseous Effluent (Note 3) Monitoring Instrumentation	H,GG
3.3.9	201	Turbine Overspeed Protection	H,GG
3.4		REACTOR COOLANT SYSTEM	
3.4.4	211	Chemistry	H,GG
3.4.8	216	Structural Integrity (Note 4)	H,GG
3.6		CONTAINMENT SYSTEMS	
3.6.1	227	Containment Leakage (Note 5)	H,GG

LCO	REPORT ITEM		PLANT
3.6.2	230	Feedwater Leakage Control	GG
3.6.7	257	Combustible Gas Control Purge System	GG
3.7		PLANT SYSTEMS	
3.7.4	266	Snubbers	H,GG
3.7.5	267	Sealed Source Contamination	H,GG
3.7.6	268	Fire Suppression Systems (6 Sections)	GG
3.7.7	269	Fire Rated Assemblies	GG
3.7.8	270	Area Temp Monitoring	GG
	271	Settlement of Class 1 Structure	H
3.7.9	272	Spent Fuel Pool Temp	GG
3.7.10	273	Flood Protection	H,GG
3.8		ELECTRICAL POWER SYSTEMS	
3.8.2	275	AC Circuits Inside Containment	H
3.8.3	276	Overcurrent Protection Devices	H,GG
3.9		REFUELING OPERATIONS	
3.9.6	286	Communications	H,GG
3.9.7	287	Refueling Equipment (3 Sections)	H,GG
3.9.10	291	Control Rod Removal (2 Sections)	H,GG
3.9.12	294	Horizontal Fuel Transfer System	GG
3.10	295	SPECIAL TEST EXCEPTIONS (Note 6)	H,GG
3.11		RADIOACTIVE EFFLUENTS (Note 3)	
3.11.1	296	Liquid Effluents	H,GG
	297	Liquid Effluents Dose	H,GG
	298	Liquid Waste Treatment	H,GG
	299	Liquid Holdup Tanks	H,GG
3.11.2	300	Gaseous Effluent Dose Rate	H,GG
	301	Gaseous Effluent Dose - Noble Gases	H,GG
	302	Gaseous Effluent Dose - Other than Noble Gas	H,GG
	303	Gaseous Radwaste Treatment	H,GG
	304	Total Dose	H,GG

BWR-TABLE 2 (Continued)

LCO	REPORT ITEM		PLANT
	305	Ventilation Exhaust Treatment System	GG
	306	Explosive Gas Mixturc	H,GG
3.11.3	308	Solid Radwaste System	H,GG
3.12		RADIOLOGICAL ENVIRONMENTAL MONITORING (Note 3)	
	309	Environmental Monitoring (3 Sections)	H,GG

Notes:

1. LCOs listed in this table may be relocated to other licensee-controlled document contingent upon NRC staff approval of the location of and controls over relocated requirements.

2. This LCO may be removed from the STS. However, if the associated Surveillance Requirement(s) is necessary to meet the OPERABILITY requirements for a retained LCC, the Surveillance Requirement(s) should be relocated to the retained LCO.

3. The staff is pursuing alternative approaches which would allow relocation of some of these LCOs on a schedule consistent with the schedule for development of the new STS. The staff is also initiating rulemaking to delete the requirement that RETS be included in Technical Specifications.

4. This LCO may be relocated out of Technical Specification. However, the associated Surveillance Requirement(s) must be relocated to Technical Specification Section 4.0, Surveillance Requirements.

5. This LCO may be rel cated, however, Pa, La, Ld and Lt must be either retained in TS or in the Bases of the appropriate containment LCO.

6. Special Test Exceptions may be included with the corresponding LCOs.

ENCLOSURE 3

POLICY ISSUE
(Information)

October 26, 1988

SECY-88-304

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: STAFF ACTIONS TO REDUCE TESTING AT POWER

Purpose: To inform the Commissioners of staff actions to reduce testing during power operation.

Background: By a staff requirements memorandum dated February 25, 1988, the Commission requested that the staff investigate the pros and cons of continuing to require surveillance and testing of equipment while the plant is at power and inform the Commission of any proposed modifications of the present requirements. In a subsequent June 20, 1988 Commission briefing on the status of the Technical Specifications Improvement Program the staff described some of its ongoing work in this area. Following that briefing the staff received another staff requirements memorandum dated July 6, 1988 requesting that a Commission paper on the results of continuing staff actions to reduce testing during power operation be provided by October 17, 1988.

Discussion: Identifying and eliminating unnecessary testing in general, and at power in particular, has long been an important objective of the staff. Beginning in 1983 with the publishing of NUREG-1024, "Technical Specifications -- Enhancing the Safety Impact," the staff initiated a program to develop analytical methods to support the implementation of changes in required surveillance intervals for testing safety-related equipment. This program was conducted by the Office of Nuclear Regulatory Research and was titled Procedures for Evaluating Technical Specifications (PETS). The effort to actually implement changes to surveillance requirements has been integrated into the current

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Technical Specifications Improvement Program associated with the Interim Commission Policy Statement on Technical Specifications Improvement issued in February 1987.

The early focus of this work has been on extending surveillance intervals for safety-related instrumentation. So far the staff has approved three topical reports which propose reduced surveillance testing of reactor protection system instrumentation, one for Westinghouse-designed pressurized water reactors and two for General Electric-designed boiling water reactors. The staff reviews of six more reports from all four reactor vendors proposing to reduce surveillance testing on reactor protection systems (RPS), engineered safety feature actuation systems (ESFAS), Emergency Core Cooling Systems (ECCS) and BWR isolation instrumentation common to RPS and ECCS are scheduled for completion this fall.

This will complete staff review of all industry proposals currently submitted to the staff for review which cover virtually all on-line testing of safety-related actuation instrumentation for major systems. Overall, when fully implemented, these changes will result in a factor of three reduction in the number of tests of these systems. The work of the PETS program was an important factor in enabling the staff to approve these changes at this time.

Other More Recent Staff Initiatives

In addition to the instrumentation work discussed above, the staff has recently broadened its efforts in this area to include major mechanical equipment and systems and to explore methods to give greater consideration to the effectiveness of maintenance programs in establishing test frequency requirements. This work was started in June of this year when NRR initiated a short-term study (approximately 120 days) of Technical Specifications testing requirements. The focus is on changes that can be implemented in a relatively short period of time and justified primarily on the basis of engineering judgment and existing or new short-term studies of actual failure rate data, as opposed to the more rigorous and time consuming PRA based analysis used to evaluate the changes in testing requirements approved for safety-related instrumentation.

The study began with a comprehensive line-by-line review of all of the testing requirements in the Technical Specifications to identify potential candidates for change. Specifications which met one or more of the following four criteria were selected for further study:

(1) The surveillance is a burden on plant personnel because the time required is not justified by the safety significance of the requirement.

(2) The surveillance could lead to a plant transient.

(3) The surveillance results in unnecessary wear to equipment.

(4) The surveillance results in exposing plant personnel to radiation levels that are not justified by the safety significance of the requirement.

An important part of the study was staff visits to five nuclear power plants to obtain information from reactor operations, maintenance, engineering, chemistry, planning, and testing personnel on which Technical Specifications surveillance requirements meet one or more of the four criteria used for the study. The sites visited were Crystal River Nuclear Plant, Unit 3; San Onofre Nuclear Generating Station, Units 1, 2, and 3; Catawba Nuclear Station, Units 1 and 2; North Anna Power Station, Units 1 and 2; and La Salle County Station, Units 1 and 2.

The study also made use of the work done as part of the NRC Nuclear Plant Aging Research (NPAR) program (NUREG-1144, Revision 1). The reports on various systems and components prepared under this program gave insight into the rate of failure of specific systems and components and also into the causes of the failures. This information was used to assess whether more testing is being done than could be justified based on the failure rates of equipment.

Findings

The technical work of the study is essentially complete and the results are being documented in a comprehensive report to be issued this month for peer review. Some of the more important general findings are summarized below. Examples of the specific recommendations that are under peer review are listed in the enclosed table. This list is not complete and it is likely that the peer review process will result in refinement to the specific recommendations.

- A large number of surveillance tests are required by the Technical Specifications. For example, the licensee for Limerick provided the following information on the total number of surveillances done on an annual basis. For 1986, with no refueling outage, 14,888 surveillances were performed. For 1987, with a refueling outage, 17,540 surveillances were performed. Approximately 98% of these were required by the Technical Specifications, the other 2% were required by other agreements between the licensee and the NRC.

  A simple averaging yields over 40 tests per day for the year with no refueling outage.

- The surveillance tests required by Technical Specifications which are the most frequent causes of reactor trips are:

  RPS Testing (PWR, BWR)
  Turbine Valve Testing (PWR, BWR)
  Control Rod Movement Testing (PWR)
  Main Steam Isolation Valve Surveillance Testing (PWR, BWR)
  Reactor Trip Breaker Testing (PWR)
  Nuclear Excore Instrumentation Testing (PWR)

- The surveillance tests required by Technical Specifications which cause the most significant equipment wear are:

  Auxiliary Feedwater Pump Testing and other safety-related pump testing in which a recirculation line is inadequately sized (PWR)
  Emergency Diesel Generator Testing

- Two programs directed by the Office of Nuclear Regulatory Research (RES) are studying ways to improve the testing of emergency diesel generators. These programs are Generic Issue B-56, "Diesel Reliability" and the Nuclear Plant Aging Research (NPAR) program. Generic Issue B-56 is scheduled for completion in June 1989. It will provide the staff with the capability to review licensee reliability programs to assure that diesel generator reliability meets the goals of the Station Blackout rule, 10 CFR 50.63, with the least adverse effect on the diesel generators.

- The surveillance tests which result in the most significant radiation dose to plant personnel are:

  Containment Purge and Exhaust Isolation Valve Leak Testing (PWRs)
  Waste Gas Storage Tank Surveillance
  Walkdowns to Verify Valve Position
  Snubber Inspections

- Surveillance and inservice testing account for approximately 20% of the annual cumulative radiation dose at a reactor. Maintenance is the largest contributor to cumulative dose.

- Improving preventive maintenance programs is an important element in reducing testing at power. A review of licensee event reports and other data shows that many of the failures found from testing are due to dirt or impurities in fluid systems, bent or broken parts, loose parts, etc., which should have been corrected before they resulted in failure. Surveillance testing can only identify that a piece of equipment is in an inoperable condition so that the time it is inoperable can be limited; preventive maintenance, however, can limit the number of failures that occur. In this way, improved preventive maintenance can make a greater contribution to reactor safety than is being made by surveillance testing.

Implementation Schedule

As noted above, some of the proposed reductions in surveillance testing for RPS and ESFAS instrumentation have already been approved with the remainder scheduled for approval before the end of the year. Individual licensees are expected to begin to submit the license amendment applications necessary to implement these changes early next year. It is possible that they could be fully implemented by the end of 1989. The implementation of these changes will result in a reduction in the frequency of tests which have been identified as being major causes of testing-induced reactor trips and thereby improve safety.

With respect to changes in testing requirements for major mechanical equipment and systems, the staff expects to complete its peer review of specific recommendations by the end of 1988. The actual implementation of the approved changes will be integrated with the implementation of the overall Technical Specifications Improvement Program through individual plant conversions to the new Standard Technical Specifications or individual license amendments. The implementation process and schedule for these types of changes at any specific plant will be based on the most cost effective use of available staff resources recognizing that, while important, they do not have the same safety significance as the changes proposed for RPS and ESFAS instrumentation.

Longer Term Activities

Based on the work that has been done to date the staff is studying the feasibility of a longer term effort with the objective of developing an entirely new approach to establishing test frequencies based on actual failure rate experience and preventive maintenance activities. Conceptually the approach would be to set minimum test intervals and reliability goals for systems and equipment and allow licensees the flexibility to increase these intervals as part of an integrated maintenance and testing program using actual failure rate history to verify that the reliability goals are being met. We understand that a similar concept is being used in Canada today. The ultimate objective would be to eliminate all testing at power for any equipment where acceptable reliability can be achieved without such testing.

A detailed schedule and milestones for this effort have not been worked out. The staff has, however, met with various industry groups and individual utilities that are pursuing programs in this area. In July of this year the staff visited the San Onofre site and met with corporate engineers and site operation and maintenance staff who are developing a program which shares many of the objectives we have established for a reliability-based integrated maintenance and surveillance program. One option for continuing this work, which is under active consideration, would be for the staff to work with an individual licensee or group of licensees to develop a pilot program to serve as a model for all plants.

The staff believes that additional work in this area could be an important first step in developing a fully integrated risk and reliability based approach to Technical Specifications.

Summary Of Conclusions: In summary, a review of operating events caused by surveillance testing shows that the large majority are caused by problems arising from surveillance on RPS and ESFAS instrumentation. However, the actual number of reactor trips related to such testing is not high. It is currently less than one per plant per year. The staff approval of the industry's proposals to increase the surveillance testing intervals for this instrumentation should, by reducing the test frequency, reduce these types of reactor trips, engineered safety features actuations, and other transients. The staff is prepared to begin to receive license amendment requests to implement these changes immediately with a goal of full implementation by the end of 1989. However, the actual rate at which changes are implemented will depend upon the extent to which individual licensees elect to participate in this voluntary program.

The implementation of the work on Technical Specifications surveillance testing of major mechanical equipment and systems will not have a large effect on reducing transients since trips due to surveillance testing make up only a small fraction of the total number of trips. Implementation of the recommendations of this work, along with the implementation of the reduction in RPS and ESFAS testing proposed in the owners groups topical reports is, however, expected to substantially reduce the number of transients caused by testing. This will result in an increase in reactor safety. The reduction in testing will also increase the performance and availability of safety-related equipment, resulting in greater reactor safety. A reduction in the Technical Specifications-related workload will result in utility technicians and engineers having more time available for other work more important to safety such as preventive maintenance.

And finally, the staff intends to continue to pursue work in developing a fully integrated risk and reliability based approach to technical specifications with the ultimate objective of eliminating all testing at power for any equipment where acceptable reliability can be achieved without such testing.

The staff plans to place a copy of this Information Paper in the Public Document Room. We will continue to keep the Commission informed of the results of this effort as they develop.

Victor Stello, Jr.
Executive Director
for Operations

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Table
Examples of recommended changes to surveillance requirements undergoing peer review

TS surveillance requirement	Recommended change
REACTIVITY CONTROL SYSTEMS	
Control rod movement testing (PWR)	Change to quarterly from every 31 days
Standby liquid control system pump test monthly (BWR)	Change surveillance test interval (STI) to quarterly
Reactor trip test to verify operability of scram discharge volume vent and drain valves. Required once every 18 months. (BWR)	Delete requirement
INSTRUMENTATION	
In core detector surveillance done weekly on CE plants and 7 days prior to use for B&W plants (PWR)	Change CE surveillance requirement to B&W surveillance requirement.
Turbine overspeed protection: Turbine valves cycled once per 7 days. Direct observation of turbine valve cycling required every 31 days (PWR, BWR)	Change all turbine valve testing to quarterly if turbine vendor agrees.
REACTOR COOLANT SYSTEM	
Leak test RCS isolation valves if in cold shutdown for more than 72 hours if not leak tested in last 9 months (PWR)	Change 72 hours to 7 days.
Check capacity of pressurizer heaters (PWR)	Change frequency to refueling intervals from every 92 days.
Demonstrate emergency power supply to pressurizer heaters is operable (done every 18 months) (PWR)	Retain for those plants where power is not from vital bus. Otherwise delete.

Table (Continued)

TS surveillance requirement	Recommended change
EMERGENCY CORE COOLING SYSTEM	
Verify boron concentration in accumulator after makeup and every 31 days (PWR)	Change to delete boron concentration check if makeup from normal source (RWST).
At least every 31 days, check for air in ECCS (PWR)	Change to after integrated leak rate test (ILRT) or maintenance on system after initial check each cycle.
Do analog channel operational test on accumulator level and pressure instrumentation (PWR)	Change to quarterly from 31 days.
CONTAINMENT	
Check areas entered in containment for loose debris after each entry (PWR)	Change to only once on last entry when successive entries are made.
Hydrogen recombiner (PWR, BWR)	Change surveillance test to refueling intervals. Presently every 6 months.
Test containment spray nozzles for obstructions every 5 years (PWR)	Extend to 10 years but require test at first refueling.
Verify operability of ice condenser doors (PWR)	Change to 18-month refueling outage for all doors rather than 25% each quarter (approved for McGuire, Catawba).
Chemical analysis of concentration of sodium tetraborate and pH of ice (PWR)	Change analysis to refueling outage (presently every 9 months)

Table (Continued)

TS surveillance requirement	Recommended change
PLANT SYSTEMS	
AFW pump surveillance test (PWR)	Change from monthly to quarterly.
Verify that control room temperature is less than specified value (typically greater than 100°F) (PWR, BWR)	Delete or revise requirement.
ELECTRICAL SYSTEMS	
Diesel generator testing (PWR, BWR)	The testing for the diesel generators should be based on reliability concepts. A reliability goal should be selected, and a program established (such as that in NUREG/CR-5078 developed for Generic Issue B-56) which will establish a testing plan to assure that the reliability goal is met.

ENCLOSURE 4

POLICY ISSUE
(Information)

October 29, 1990

SECY-90-366

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: REPORT ON THE STATUS OF THE TECHNICAL SPECIFICATIONS IMPROVEMENT PROGRAM

Purpose: To provide the Commission with an update on the current status of the Technical Specifications Improvement Program.

Summary: The staff has previously briefed the Commission on the status of the Technical Specifications Improvement Program. At the last briefing the staff told the Commission that it expected the new standard technical specifications to be completed by April 1990. Several unanticipated problems have prevented the industry and the staff from meeting this schedule: (1) The number of changes proposed by the industry was greater than anticipated, and (2) a very large and time-consuming word processing and editing effort has been required.

The staff expects to complete the development of the new standard technical specifications and present the results to ACRS before the end of 1990. A complete draft will be ready in November 1990. A review and approval process will then take several more months to complete. The staff now expects to complete work on the new standard technical specifications in spring 1991. The staff and the industry groups (the owners groups and NUMARC) are all giving high priority to completion of the new Standard Technical Specifications.

Background: Because the Technical Specifications Improvement Program is a major NRC initiative, the staff has briefed the Commission several times on the status of this program. This paper provides yet another update on the staff and the industry effort to bring this program to fruition.

On February 6, 1987, the Commission issued the interim Policy Statement on technical specifications improvement. This document served as the basis for identifying improvements to be made to the existing standard technical specifications (STS). It

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specified criteria to be used to decide which requirements were to be retained in the technical specifications and which requirements were to be relocated to licensee-controlled documents. It also called for a strong program to implement 10 CFR 50.59 requirements for those items relocated from the technical specifications. Using these criteria, on May 9, 1988, after discussions with the industry, the staff issued letters to the owners groups listing those specifications to be relocated from the STS and those to remain. Based on the guidance of these letters, the owners groups prepared and submitted to the staff proposed new STS. These proposed new STS not only reflected the policy of relocating requirements that did not meet the criteria of the interim Policy Statement but also were written in an improved format from a human factors viewpoint. In addition, the owners groups' submittals contained numerous substantive technical changes that were not part of the original plan for the Technical Specifications Improvement Program.

Throughout this process, the staff briefed the Commission several times. At the most recent briefing, on June 2, 1989, the staff gave the Commission the dates for each owners group submittal and the date the staff anticipated producing the safety evaluation report (SER) for each submittal. The safety evaluations for the new standard technical specifications were to be issued no later than spring 1990.

Since the June 2, 1989, briefing, the staff revised the original schedule.

This paper provides the Commission with the current status of the Technical Specifications Improvement Program, and in particular, the progress made to date and the current schedule for completion.

Discussion: The staff now plans to complete its review of the five sets of new STS in the spring of 1991. A complete draft for each set will be ready in November 1990. This has been a major staff effort. There are currently 15 members in the Technical Specifications Branch, one senior reactor operator instructor (a foreign-assignee working with the branch), approximately 20 technical experts in other branches (on a part-time basis), and approximately 10 contractors working on the review.

The staff has reviewed approximately 4,100 proposed changes to the technical specifications, held approximately 90 meetings with the owners groups to discuss these changes, and is now preparing approximately 13,000 pages of written text which will comprise the 5 sets of the new STS. A number of these pages are changed and have required retyping several times as a result of continuing discussions between the staff and the owners groups. The staff, through contractors, is doing all the word processing and editorial work as well as the technical review.

The staff evaluated operator acceptance of the new STS at the NRC Technical Training Center simulator in Chattanooga. (The operators enthusiastically accepted the new STS). The staff also performed its own major review of surveillances required by the technical specifications. The results of this study are incorporated in the new STS and will also be issued to the industry as a line-item improvement. As a parallel effort, as directed by the Commission, the staff is developing guidelines for reviews conducted by licensees under 10 CFR 50.59. Following the NRC staff review, the industry issued a report (NSAC-125) which provides guidance on the performance of reviews required by 10 CFR 50.59. Working with the industry, members of the Technical Specifications Branch briefed all five regions on the work done to date on these 10 CFR 50.59 guidelines.

The staff has also completed its review of all limiting conditions for operation (LCOs) and surveillance requirements. The last major effort, the review of the bases, is now nearing completion. This review has required a large amount of rewriting but should be completed within the next month.

Before reaching agreement on the various technical issues, the staff has held lengthy discussions with the industry. These efforts have been very productive in reducing the number of open issues. However, some open issues will remain between the staff and industry at the time the staff publishes the complete draft STS for comment. These residual open issues will continue to be addressed during the period of public ACRS and CRGR review.

A lead plant from each owners group has been participating in the review of the new STS. The purpose of this participation is to validate the new STS for that plant, that is, to obtain assurance that the generic STS can effectively be applied to an operating reactor of that design.

Following the completion of the generic new STS and the validation effort, the review of the application of the new STS to each of the lead plants will be completed. The staff anticipates that this task will require several months after the work on the new STS is finished.

In summary, because of (1) the large number of technical issues to be resolved that were not originally anticipated, and (2) the large volume of clerical (word processing and editing) work to be completed, the staff has had to revise the schedule originally provided to the Commission. The staff has nearly completed the review of the new STS for each owners group. In November 1990, drafts (for each owners group) of the new STS are scheduled to be completed. The staff expects to resolve any public comment, complete ACRS and CRGR review and publish the final versions of the new STS in the spring of 1991.

Throughout this effort, the staff has emphasized producing a high quality product. The industry also shares this view. With the task of producing the new STS close to completion, the staff will take the time required to ensure that the final product will be of high quality.

James M. Taylor
Executive Director
for Operations

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