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.

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

cc see next page

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(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

cc w/encl:

Mr. Robert Gill
B&W Owners Group
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Mr. R. E. Bradley
BWR Owners Group
c/o Georgia Power
Nuclear Operations Department
14th Floor
333 Piedmont Avenue
Atlanta, Georgia 30308

Mr. Edward Lozito
Westinghouse Owners Group
c/o Virginia Power
P. O. Box 26666
Richmond, Virginia 23261

Mr. Joseph B. George
Westinghouse Owners Group
Texas Utilities
400 North Olive
Dallas, Texas 75201

Mr. Stewart Webster
CE Owners Group
1000 Prospect Hill Road
Winstor, Connecticut 06095-0500

Mr. R. A. Bernier
CE Owners Group
c/o Arizona Nuclear Power Project
P. O. Box 52034
M.S. 7048
Phoenix, Arizona 85072

Mr. Thomas Tipton
NUMARC
1776 Eye Street, N.W.
Suite 300
Washington, D. C. 20006-2496

Identical Letters mailed to the following:

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

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
BWR Owners Group
c/o Commonwealth Edison Company
Room 34FN East
P. O. Box 767
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 <u>only</u> instrumentation used to detect actual leaks and <u>not</u> 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

B&W-TABLE 1 (Continued)

LCO		CRITERIA
3.4.7.2	Operational Leakage	2
3.4.9	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)

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 -- [illegible]ld. ≥ 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

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

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

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

BWR-TABLE 2 (Continued)

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 Mixture	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 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 Specification. 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.

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

Contact:
Edward J. Butcher, NRR
49-21183

8811070008

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

- o 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.

- o 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

Enclosure:
As stated

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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)

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

CONTACT: Richard M. Lobel, OTSB, NRR
x21185

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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
James M. Taylor
Executive Director
for Operations

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WASHINGTON, D. C. 20555

DEC 1 1 '91

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

FROM: Robert M. Bernero, Director
Office of Nuclear Material Safety
and Safeguards

SUBJECT: TECHNICAL POSITION ON WASTE FORM REVISION 1

Enclosed is a draft revision (Rev. 1) to the Technical Position (TP) on Waste Form (Enclosure 1). The revision consists primarily of a new appendix (Appendix A) that addresses the use of cement for the solidification and stabilization of Class B and Class C low-level radioactive waste. This proposed revision of the TP on Waste Form is the first to be initiated since the TP was issued in May 1983.

The TP revision focuses on the requirement, contained in 10 CFR 61.56(b), that low-level radioactive wastes possess long-term (e.g., 300-year) structural stability. Low-Level Waste (LLW) generators must certify, in accordance with requirements in 10 CFR 20.311, that their wastes satisfy the waste form requirements in Part 61. The TP is intended to give guidance to waste generators and processors on ways that reasonable assurance can be provided that the wastes will possess the long-term structural stability required by Part 61. Under an accord reached in 1983 with the sited Agreement States, the State authorities (in Nevada, South Carolina, and Washington) agreed to continue to permit the disposal of cement-solidified wastes at their LLW disposal facilities, while the Office of Nuclear Material Safety and Safeguards staff reviewed vendor-developed formulations under a topical report review program. In effect, the cement-solidified Class B and C waste forms were "grandfathered," pending the outcome of the staff reviews. Staff has to this time, however, not approved any commercial LLW cement formulations due to the fact that current guidance does not incorporate existing technical information. Updated guidance will provide a firm basis for requesting additional information necessary to resolve all presently known technical concerns.

There have been a number of incidents involving cement-solidified waste forms that have not solidified properly. These incidents, supplemented by laboratory test results, indicate that some, as yet unquantified, fraction of the cement-solidified LLW currently being placed in LLW disposal facilities may not be in compliance with Part 61 stability requirements. It is imperative, therefore, that the nuclear industry and NRC staff have adequate technical guidance to enable well-founded and supportable judgments to be made of the ability of cement-solidified LLW forms to meet the stability requirements of Part 61. The revised TP would end the grandfathering of cement-solidified LLW and provide a justifiable basis for decisions to be made on cement waste form acceptability.

The Low-Level Radioactive Waste Policy Act of 1980 as amended calls for the

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establishment of a national program with a regulatory framework that is applicable to all waste generators and disposal facilities without regard to cost/benefit or backfit considerations. Therefore, the proposed revision to the TP would be applicable to reactor licensees, nuclear material licensees and disposal facilities licensees.

The current situation is the same as that which existed in 1983 when the TP was first promulgated. At that time the Committee to Review Generic Requirements (CRGR) was briefed on the TP and suggested three items be considered in the development of LLW TP's:

1. TP's should be forwarded to the Advisory Committee on Reactor Safeguards (ACRS) and published for further public comment with special efforts to obtain comments from non-power reactor licensees.

2. A letter should be prepared to accompany the TP that is coordinated with all affected program offices.

3. In developing and implementing waste requirements and guidance, the staff should closely coordinate activities with State and local governments.

The above suggestions, made by the CRGR on the 1983 TP, have all been attended to as follows for the proposed Revision 1:

Item 1: The draft TP was forwarded to the Advisory Committee on Nuclear Waste (ACNW) with a follow-up meeting in August. The meeting agenda item was noticed in the Federal Register. Copies of the draft TP were provided to vendors, reactor licensees and representative groups such as the Electric Power Research Institute (EPRI), the Nuclear Management and Resources Council (NUMARC), and the Edison Electric Institute (EEI) with requests for comments. A meeting was held at NRC Headquarters with these groups to discuss the draft TP revision. Comments received from the ACNW (Enclosure 2) and others have been factored into the current draft of the TP.

Item 2: Affected program offices, Office of State Programs (OSP), Office of Nuclear Reactor Regulation (NRR), and Office of the General Counsel (OGC) were provided copies of the draft TP and asked for comments. They have expressed their support for the TP, verbally and/or in writing (see Enclosure 3).

Item 3: We have, as noted above, worked closely with the Agreement State authorities in developing the draft guidance. This interaction included a discussion of the TP and related waste form matters in an Agreement State Workshop, which was co-sponsored by OSP and NMSS and held in Bethesda in June. Copies were provided to the State authorities following the June Workshop with a request for comments. Though the States expressed their support verbally at the Workshop, they have not provided written comments on the TP to date. Before the provisions in the draft TP are implemented, further interactions with the States will be carried out to obtain their input and agreement for the scheduling of implementation of key effects of the revision, such as the ending of the grandfathering of cement-solidified LLW.

In addition to the 1983 CRGR meeting, a briefing of the CRGR was held on September 22, 1988, to provide the status of NMSS waste form activities. As reflected in the minutes of the 147th CRGR Meeting (see Enclosure 4), the Committee requested to be kept informed regarding the status of the LLW topical report reviews, and agreed that CRGR did not have to routinely review staff actions in this area. The current revision falls into the same category as the initial 1983 TP and thus does not require the review by the CRGR. In accordance with your report (on the contents of packages submitted to CRGR), we are, however, forwarding for your information the enclosed materials.

For the reasons specified above, we are anxious to proceed with the release and implementation of the TP revision as soon as possible. The intent is to release the final TP revision in early 1991 (following the Office of Management and Budget (OMB) review) and implement the provisions as soon as practical thereafter. The method of release will be a Federal Register Notice and a transmittal letter to all NRC licensees and Agreement States. The letter will explain the implementation dates and details. We request your support in this endeavor. If the CRGR should have any further need for additional information, the NMSS point of contact on this matter is Dr. Michael Tokar.

Robert M. Bernero, Director
Office of Nuclear Material Safety
and Safeguards

Enclosures:
1. Draft Revision, Technical Position on Waste Form
2. Ltr from Moeller (ACNW) to Chairman Carr, dated 9/6/90
3. Ltr from Treby (OGC) to Bangart (NMSS), dated 6/18/90
4. Minutes of CRGR Meeting Number 147, Jordan to Stello, dated 10/15/88

DRAFT

United States Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Washington, D.C. 20555

TECHNICAL POSITION

ON

WASTE FORM

(Revision 1)

DRAFT

Prepared by: Technical Branch
Division of Low-Level Waste Management
and Decommissioning

July 1990

Enclosure 1

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Technical Position on Waste Form

Appendix A - Cement Stabilization

Technical Position on Waste Form

A. INTRODUCTION

The regulation, "Licensing Requirements for Land Disposal of Radioactive Waste," 10 CFR Part 61, establishes a waste classification system based on the radionuclide concentrations in the wastes. Class B and C waste are required to be stabilized. Class A wastes have lower concentrations and may be segregated without stabilization. Class A wastes may also be stabilized and disposed of with stabilized Class B and C wastes. All Class A liquid wastes, however, require solidification or absorption to meet the free liquid requirements. Structural stability is intended to ensure that the waste does not degrade and (a) promote slumping, collapse, or other failure of the cap or cover over a near-surface disposal trench and thereby lead to water infiltration, or (b) impart a substantial increase in surface area of the waste form that could lead to an increase in leach rate. Stability is also a factor in limiting exposure to an inadvertent intruder since it provides greater assurance that the waste form will be recognizable and nondispersable during its hazardous lifetime. Structural stability of a waste form can be provided by the waste form itself (as with activated stainless steel components), by processing the waste to a stable form (e.g., solidification), or by emplacing the waste in a container or structure that provides stability (e.g., high integrity container or engineered structure).

This technical position on waste form was initially developed in 1983 to provide guidance to both fuel-cycle and non-fuel-cycle waste generators on waste form test methods and results acceptable to the NRC staff for implementing the 10 CFR Part 61 waste form requirements. It has been used as an acceptable approach for demonstrating compliance with the 10 CFR Part 61 waste stability criteria. This position includes guidance on (1) the processing of wastes into an acceptable, stable waste form, (2) the design of acceptable high integrity containers, (3) the packaging of filter cartridges, and (4) minimization of radiation effects on organic ion-exchange resins. The regulation, 10 CFR 20.311, requires waste generators and processors to certify that their waste forms meet the requirements of Part 61 (including the requirements for structural stability). The recommendations and guidance provided in this technical position are an acceptable method to provide such certification by waste generators. One way of demonstrating conformance with the general recommendations contained in this technical position is to reference an approved Topical Report, because such reports are reviewed and approved in accordance with the acceptance criteria contained in this technical position. Additional actions (e.g., plant-specific process control procedures) by waste generators, however, to demonstrate that a stabilized plant-specific waste stream satisfies Part 61 waste form requirements, will be needed.

Since the initial conception of the Technical Position, it has been the intent of the NRC staff to provide additional guidance on waste form as it became necessary to address other pertinent waste form issues. One such issue involves the use of cement to stabilize low-level wastes. Field experience and laboratory testing of cement-solidified low-level radioactive waste has indicated that some unique chemical and physical interactions can occur between the cement constituents and the chemicals and compounds that can exist in the waste materials. Therefore, an appendix (Appendix "A") dealing with the qualification testing, performance confirmation and reporting of mishaps involving cement-stabilized waste forms has been included in this revision to the Technical Position.

To provide more comprehensive guidance on cement stabilization of low-level radioactive waste, Appendix A addresses several areas of concern that were not considered in the May 1983, Revision 0, version of this Technical Position. Thus, information and guidance on cement waste form specimen preparation, statistical sampling and analysis, waste characterization, process control program (PCP) specimen preparation and examination, surveillance specimens and reporting of mishaps are provided in Appendix A. The guidance provided in Appendix A is the culmination of an extended period of study and information gathering and exchange between the NRC staff and representatives of various sectors of the nuclear industry, including government laboratories, cement processing vendors, other waste form vendors, nuclear utilities, state regulatory agencies, and industry representative organizations such as the Nuclear Management Resources Council (NUMARC) and the Electric Power Research Institute (EPRI). Especially useful in the development of the guidance in Appendix A was the information exchanged in a Workshop on Cement Stabilization of Low-Level Radioactive Waste (Ref. 1).

B. BACKGROUND

Historically, waste form and container properties were considered of secondary importance to good site selection; a properly operated site having good geologic and hydrologic characteristics was considered the only barrier necessary to isolate low-level radioactive wastes from the environment. As experience in operating low-level waste disposal sites was acquired, however, it became apparent that the waste form should play a significant role in the overall plan for managing these wastes.

The regulation for near-surface disposal of radioactive wastes, 10 CFR Part 61, includes requirements which must be met by a waste form to be acceptable for near-surface disposal. The regulation includes a waste classification system which divides waste into three general classes: A, B, and C.

The classification system is based on the overall disposal hazards of the wastes. Certain minimum requirements must be met by all wastes. These minimum requirements are presented in Section 61.56(a) and involve basic packaging criteria, prohibitions against the disposal of pyrophoric, explosive, toxic and infectious materials, and requirements to solidify or absorb liquids.

In addition to the minimum requirements, Class B and C wastes are required to have structural stability. As stated in Section 61.56(b) of the rule, stability requires that the waste form maintain its structural integrity under the expected disposal conditions. Structural stability is necessary to inhibit (a) slumping, collapse, or other failure of the disposal trench (if an engineered structure is not used) resulting from degraded wastes which could lead to water infiltration, radionuclide migration, and costly remedial care programs and (b) radionuclide release from the waste form that might ensue due to increases in leaching that could be caused by premature disintegration of the waste form. Stability is also considered in the intruder pathways where it is assumed that wastes are recognizable after the active control period, and that, therefore, continued inadvertent intrusion would be unlikely. To the extent practical, Class B and C waste forms should maintain gross physical properties and identity over a 300 year period.

To ensure that Class B and C wastes will maintain stability, the following conditions should be met:

a. The waste should be a solid form or in a container or structure that provides stability after disposal.

b. The waste should not contain free standing and corrosive liquids. That is, the wastes should contain only trace amounts of drainable liquid, and, as required by 10 CFR 61.56(b)(2), in no case may the volume of free liquid exceed one percent of the waste volume when wastes are disposed of in containers designed to provide stability, or 0.5 percent of the waste volume for solidified wastes.

c. The waste or container should be resistant to degradation caused by radiation effects.

d. The waste or container should be resistant to biodegradation.

e. The waste or container should remain stable under the compressive loads inherent in the disposal environment.

f. The waste or container should remain stable if exposed to moisture or water after disposal.

g. The as-generated waste should be compatible with the solidification medium or container.

A large portion of the waste produced in the nuclear industry, including waste from nuclear power plants, is in a form which is either liquid or in a wet solid form (e.g., resins, filter sludge, etc.) and requires processing to achieve an acceptable form for burial. The wet wastes, regardless of their classification, are required to be either absorbed or solidified. To assure that this processing will consistently produce a product which is acceptable for disposal and will meet disposal site license conditions, nuclear power plant licensees are required to process their wastes in accordance with a plant-specific process control program (PCP). Guidance for such PCPs was provided in NRC Standard Review Plan Section 11.4, "Solid Waste Management Systems," <u>NUREG-0800</u> (Ref. 2) and its accompanying Branch Technical Position ETSB 11-3, "Design Guidance for Solid Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants," (revised in July 1981). However, 10 CFR Part 61 became effective in January 1983, providing requirements regarding waste form, and superseding certain of the guidance previously provided in NUREG-0800. Licensee's PCPs provide assurance that the processing of wet radioactive wastes will result in waste forms that meet the requirements of 10 CFR Part 61 and low-level waste disposal sites licenses. Plant-specific PCPs developed and approved without consideration of Part 61 should be revised to provide assurance that applicable Part 61 requirements will be satisfied. In many cases, licensee PCPs are based on generally applicable (generic) PCPs contained in vendor-submitted topical reports that are reviewed by the NRC for referencing in licensing actions.

The guidance in this technical position may also serve as the basis for qualifying generic PCPs for Class B and C wastes. Applicable generic test data (e.g., topical reports) may be used for generic PCP qualification, and may be used in part as the basis for a plant-specific PCP. PCPs for solidified Class A waste products that are to be segregated from Class B and C wastes need only demonstrate that the product is a free-standing monolith with no more than 0.5 percent of the waste volume as free liquid.

An alternative to processing some Class B and C waste streams, particularly ion exchange resins and filter sludges, is the use of a high integrity container (HIC). The high integrity container would be used to provide the long-term stability required to meet the structural stability requirements in 10 CFR Part 61. The design of the high integrity container should be based on its specific intended use in order to ensure that the waste contents, as well as interim storage and ultimate disposal environments, will not compromise its integrity over the long-term. As with waste solidification, a PCP for dewatering wet solids in HICs or liners should be developed and utilized to ensure that the free liquid requirements in 10 CFR Part 61 are being met.

C. REGULATORY POSITION

1. Solidified Class A Waste Products

a. Solidified Class A waste products which are segregated from Class B and C wastes should be free standing monoliths and have no more than 0.5 percent of the waste volume as free liquids as measured using the method described in ANS 55.1 (Ref. 4).

b. Class A waste products which are not segregated from Class B and C wastes should meet the stability guidance for Class B and C wastes provided below.

2. Stability Guidance for Processed (i.e., Solidified) Class B and C Wastes

The stability guidance in this technical position for processed wastes should be implemented through the qualification of the individual licensee's PCP. Generic test data may be used for qualifying generic PCPs, and incorporated as part of the individual licensee's (i.e., plant-specific) PCP. Tests to demonstrate waste form stability through a generic testing program include the following:

a. Solidified waste specimens should have compressive strengths of at least 60 psi when tested in accordance with ASTM C39 (Ref. 5). Compressive strength tests for bituminous products should be performed in accordance with ASTM D1074 (Ref. 6).

Many solidification agents (such as cement) will be easily capable of meeting the 60 psi limit for properly solidified wastes. For such cases, process control parameters should be developed to achieve maximum practical compressive strengths, not simply to achieve the minimum acceptable compressive strength; (see Section II.B of Appendix A for further guidance on cement-stabilized wastes).

b. Waste specimens should be resistant to thermal degradation. The heating and cooling chambers used for the thermal degradation testing should conform to the description given in ASTM B553, Section 3 (Ref. 7). Samples suitable for performing compressive strength tests in accordance with ASTM C39 or ASTM D1074 should be used. Samples should be placed in the test chamber and a series of 30 thermal cycles carried out in accordance with Section 5.4.1 through 5.4.4 of ASTM B553. The high temperature limit should be 60°C and the low temperature limit -40°C. Following testing the waste specimens should have the maximum practical compressive strengths; (a minimum compressive strength of 60 psi as tested using ASTM D1074 is acceptable for bituminized waste forms--for cement-stabilized wastes see Section II.C of Appendix A).

c. The specimens for each proposed waste stream formulation should remain stable after being exposed in a radiation field equivalent to the maximum level of exposure expected from the proposed wastes to be solidified. Specimens for each proposed waste stream formulation should be exposed to a minimum of 10E+8 Rads in a gamma irradiator or equivalent. If the maximum level of exposure is expected to exceed 10E+8 Rads, testing should be performed at the expected maximum accumulated dose. Following irradiation the irradiated specimens should have the maximum practical compressive strengths (a minimum compressive strength of 60 psi as tested using ASTM D1074 is acceptable for bituminized waste forms--for cement-stabilized wastes see Appendix A).

d. Specimens for each proposed waste stream formulation should be tested for resistance to biodegradation in accordance with both ASTM G21 and ASTM G22 (Refs. 8 & 9, respectively). No indication of culture growth should be visible. Specimens should be suitable for compression testing in accordance with ASTM C39 or ASTM D1074, as applicable. Following the biodegradation testing, specimens should have the maximum practical compressive strengths (a minimum compressive strength of 60 psi as tested using ASTM D1074 is acceptable for bituminized waste forms--see Section II.E of Appendix A for guidance on biodegradation testing of cement-stabilized wastes).

For polymeric or bitumen products, some visible culture growth from contamination, additives, or biodegradable components on the specimen surface that does not relate to overall substrate integrity may be present. For these cases, additional testing should be performed. If culture growth is observed upon completion of the biodegradation test for polymeric or bitumen products, the test specimens should be removed from the culture and washed free of all culture and growth with water, with only light scrubbing. An organic solvent compatible with the substrate may be used to extract surface contaminants. The specimen should be air dried at room temperature and the test repeated. Specimens should have observed culture growths rated no greater than 1 in the repeated ASTM G21 test. The specimens should have no observed growth in the repeated ASTM G22 test. Compression testing should be performed in accordance with ASTM C39 or ASTM D1074, as applicable, following the repeated G21 and G22 tests. The minimum acceptable compressive strength for bituminized waste forms is 60 psi. Maximum practical compressive strengths should be established for other media.

If growth is observed following the extraction procedure, longer term testing of at least six months should be performed to determine biodegradation rates. The Bartha-Pramer Method (Ref. 10) is acceptable for this testing. Soils used should be representative of those at burial grounds. Biodegradation extrapolated for full-size waste forms to 300 years should produce less than a 10 percent loss of the total carbon in the waste form.

e. Leach testing should be performed for a minimum of 90 days (5 days for cement-stabilized waste forms--see Section II.F of Appendix A for cement-stabilized wastes) in accordance with the procedure in ANS 16.1 (Ref. 11). Specimen sizes should be consistent with the samples prepared for the ASTM C39 or ASTM D1074 compressive strength tests. In addition to the demineralized water test specified in ANS 16.1, additional testing using other leachants specified in the Standard should also be performed to confirm the solidification agents leach resistance in other leachant media. It is preferred that the synthesized sea water leachant also be tested. In addition, it is preferable that radioactive tracers be utilized in performing the leach tests. For proposed nuclear power station waste streams, cobalt, cesium, and strontium should be used as tracers. The leachability index, as calculated in accordance with ANS 16.1, should be greater than 6.0.

f. Waste specimens should maintain maximum practical compressive strengths as tested using ASTM C39 or ASTM D1074, following immersion for a minimum period of 90 days. Immersion testing may be performed in conjunction with the leach testing; (see Section II.G of Appendix A for guidance on cement-stabilized wastes).

g. Waste specimens should have less than 0.5 percent by volume of the waste specimen as free liquids as measured using the method described in ANS 55.1. Free liquids should have a pH between 4 and 11; (for cement-solidified water, free liquids should have a minimum pH of 9--see Section II.H of Appendix A).

h. If small, simulated laboratory size specimens are used for the above testing, test data from sections or cores of the anticipated full-scale products should be obtained to correlate the characteristics of actual size products with those of simulated laboratory size specimens. This testing may be performed on non-radioactive specimens. Correlation testing should be performed using 90-day immersion (including post-immersion compression) tests on the most conservative waste stream(s) intended for use for the particular solidification medium; i.e, the waste stream that presents the most difficulty in consistently producing a stable product(s). For cement-solidified waste forms, the mixed bead resin waste stream is expected to be the most conservative. For bituminized wastes, the sodium sulfate waste stream should be used. The full-scale specimens should be fabricated using solidification equipment the same as or comparable to that used for processing actual low-level radioactive wastes in the field.

i. Waste samples from full-scale specimens should be destructively analyzed to ensure that the product produced is homogeneous to the extent that all regions in the product can expect to have compressive strengths representative of the compressive strength as determined by testing lab-scale specimens (i.e., that meet the criteria called out in Section C2.a. above). Full-scale specimens may be fabricated using simulated non-radioactive products; however, the specimens should be fabricated using solidification equipment that is the same as or comparable to that used in the field for actual low-level radioactive wastes.

3. Radiation Stability of Organic Ion-Exchange Resins

To ensure that organic ion exchange resins will not undergo adverse degradation effects from radiation, resins should not be generated having loadings that will produce greater than 10E+8 Rads total accumulated dose. For Cs-137 and Sr-90 a total accumulated dose of 10E+8 Rads is approximately equivalent to a 10 Ci/ft concentration in resins in the unsolidified, as-generated form. In the event that the waste generator considers it necessary to load resins higher than 10E+8 Rads, it should be demonstrated that the specific resin will not undergo radiation degradation at the proposed higher loading. The test method should adequately simulate the chemical and radiologic conditions expected. A gamma irradiator or equivalent should be utilized for these tests. There should be no adverse swelling, acid formation or gas generation that will be detrimental to the proposed final waste product.

4. High Integrity Containers

a. The maximum allowable free liquid in a high integrity container should be less than one percent of the waste volume as measured using the method described in ANS 55.1 A process control program should be developed and qualified to ensure that the free liquid requirements in 10 CFR Part 61 will be met upon delivery of the wet solid material to the disposal facility. This process control program qualification should consider the effects of transportation on the amount of drainable liquid which might be present.

b. High integrity containers should have as a design goal a minimum lifetime of 300 years. The high integrity container should be designed to maintain its structural integrity over this period.

c. The high integrity container design should consider the corrosive and chemical effects of both the waste contents and the disposal environment. Corrosion and chemical tests should be performed to confirm the suitability of the proposed container materials to meet the design lifetime goal.

d. The high integrity container should be designed to have sufficient mechanical strength to withstand horizontal and vertical loads on the container equivalent to the depth of proposed burial assuming a cover material density of 120 lbs/ft^3. The high integrity container should also be designed to withstand the routine loads and effects from the waste contents, waste preparation, transportation, handling, and disposal site operations, such as trench compaction procedures. This mechanical design strength should be justified by conservative design analyses.

e. For polymeric material, design mechanical strengths should be conservatively extrapolated from creep test data. It should be demonstrated for high integrity containers fabricated from polymeric materials that the containers will not undergo tertiary creep, creep buckling, or ductile-to-brittle failure over the design life of the containers.

f. The design should consider the thermal loads from processing, storage, transportation and burial. Proposed container materials should be tested in accordance with ASTM B553 in the manner described in Section C2(b) of this technical position. No significant changes in material design properties should result from this thermal cycling.

g. The high integrity container design should consider the radiation stability of the proposed container materials as well as the radiation degradation effects of the wastes. Radiation degradation testing should be performed on proposed container materials using a gamma irradiator or equivalent. No significant changes in material design properties should result following exposure to a total accumulated dose of 10 E+8 Rads. If it is proposed to design the high integrity container to greater accumulated doses, testing should be performed to confirm the adequacy of the proposed materials. Test specimens should be prepared using the proposed fabrication techniques.

High integrity container designs using polymeric materials should also consider the effects of ultra-violet radiation. Testing should be performed on proposed materials to show that no significant changes in material design properties occur following expected ultra-violet radiation exposure.

h. The high integrity container design should consider the biodegradation properties of the proposed materials and any biodegradation of wastes and disposal media. Biodegradation testing should be performed on proposed container materials in accordance with ASTM G21 and ASTM G22. No indication of culture growth should be visible. The extraction procedure described in Section C2(d) of this technical position may be performed where indications of visible culture growth can be attributable to contamination, additives, or biodegradable components on the specimen surface that do not affect the overall integrity of the substrate. It is also acceptable to determine biodegradation rates using the Bartha-Pramer Method described in Section C2(d). The rate of biodegradation should produce less than a 10 percent loss of the total carbon in the container material after 300 years. Test specimens should be prepared using the proposed material fabrication techniques.

i. The high integrity container should be capable of meeting the requirements for a Type A package as specified in 49 CFR 173.411 and 173.412. Conditions that may be encountered during transport or movement are to be addressed by meeting the requirements of 10 CFR 71.71. j. The high integrity container and the associated lifting devices should be designed to withstand the forces applied during lifting operations. As a minimum the container should be designed to withstand a 3g vertical lifting load.

k. The high integrity container should be designed to avoid the collection or retention of water on its top surfaces in order to minimize accumulation of trench liquids which could result in corrosive or degrading chemical effects.

l. High integrity container closures should be designed to provide a positive seal for the design lifetime of the container. The closure should also be designed to allow inspections of the contents to be conducted without damaging the integrity of the container. Passive vent designs may be utilized if needed to relieve internal pressure. Passive vent systems should be designed to minimize the entry of moisture and the passage of waste materials from the container.

m. Prototype testing should be performed on high integrity container designs to demonstrate the container's ability to withstand the proposed conditions of waste preparation, handling, transportation and disposal.

n. High integrity containers should be designed, fabricated, and used in accordance with a quality assurance program. The quality assurance program should address the following topics concerning the high integrity container: fabrication, testing, inspection, preparation for use, filling, storage, handling, transportation, and disposal. The quality assurance program should also address how wastes which are detrimental to high integrity container materials will be precluded from being placed into the container. Special emphasis should be placed on fabrication process control for those high integrity containers which utilize fabrication techniques such as polymer molding processes.

5. Filter Cartridge Wastes

For Class B and C wastes in the form of filter cartridges, the waste generator should demonstrate that the selected approach for providing stability will meet the requirements in 10 CFR Part 61. Encapsulation of the filter cartridge in a solidification binder or the use of a high integrity container are acceptable options for providing stability. When high integrity containers are used, waste generators should demonstrate that protective means are provided to preclude container damage during packaging handling and transportation.

6. Reporting of Mishaps

In all future reviews and approvals of stabilization media and high integrity containers, waste generators, vendors and processors will, as a condition of approval, be asked to commit to reporting any knowledge they may have of misuse or failure of their waste forms and containers. Such mishaps include, but are not necessarily limited to, the following:

a. The failure of high integrity containers used to ensure structural stability. Such failure may be evidenced by changed container dimensions, cracking, or injury from mishandling (e.g., dropping or impacting against another object).

b. The misuse of high integrity containers, as evidenced by a quantity of free liquid greater than one percent of container volume, or an excessive void space within the container; (such use is in violation of 10 CFR 61.56(a)).

c. The production of a solidified Class B or C waste form that has any of the following characteristics;

1. greater than 0.5 percent volume of free liquid.

2. concentrations of radionuclides greater than the concentrations demonstrated to be stable in the waste form in qualification testing accepted by the regulatory agency.

3. greater or lessor amounts of solidification media than were used in qualification testing accepted by the regulatory agency.

4. contains chemical ingredients not present or accounted in qualification testing accepted by the regulatory agency.

5. shows instability evidenced by crumbling, cracking, spalling, voids, softening, disintegration, nonhomogeneity, or change in dimensions.

6. evidences processing phenomena that exceed the limiting processing conditions identified in applicable topical reports or process control programs, such as foaming, excessive temperature, premature or slow hardening, production of volatile material, etc.

Waste form mishaps should be reported to the NRC's Director of the Division of Low-Level Waste Management and Decommissioning and the designated State disposal site regultory authority within 30 days of knowledge of the incident. For any such waste form mishap occurrence, the affected waste form should not be shipped off-site until approval is obtained from the disposal site regulatory authority. The reason for this is that the low-level waste generators and processors are required by 10 CFR 20.311 to certify that their waste forms meet all applicable requirements of 10 CFR Part 61, and waste forms that are subject to the types of mishaps mentioned above may not possess the required long-term structural stability. When mishaps of the nature described above occur, it is expected that, before the waste form is shipped to a disposal facility, either adequate mitigation of the potential effects on the waste form or an acceptable justification concerning the lack of any potential significant effects of the affected waste form on the overall performance of the disposal facility would be provided.

D. IMPLEMENTATION

This technical position reflects the current NRC staff position on acceptable means for meeting the 10 CFR Part 61 waste stability requirements. Therefore, except in those cases in which the waste generator, vendor, and/or processor proposes an acceptable alternative method for complying with the stability requirements of 10 CFR Part 61, the guidance described herein will be used in the evaluation of the acceptability of waste forms for disposal at near-surface disposal facilities.

E. REFERENCES

1. "Proceedings of the Workshop on Cement Stabilization of Low-Level Radioactive Waste," NUREG/CP-0103, October 1989.

2. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (LWR Edition), NUREG-0800, July, 1981.

3. "Update on Waste Form and High Irtegrity Container Topical Report Review Status, Identification of Problems with Cement Solidification, and Reporting of Waste Mishaps," NRC Information Notice No. 90-xx, (in preparation).

4. ANS 55.1, "American National Standard for Solid Radioactive Waste Processing System for Light Water Cooled Reactor Plants," American Nuclear Society, 1979.

5. ASTM C39, "Compressive Strength of Cylindrical Concrete Specimens," American Society for Testing and Materials, 1979.

6. ASTM D1074, "Compression Strength of Bituminous Mixtures," American Society for Testing and Materials, 1980. 7. ASTM B553, "Thermal Cycling of Electroplated Plastics," American Society for Testing and Materials, 1979.

8. ASTM G21, "Determining Resistance of Synthetic Polymeric Materials to Fungi," American Society for Testing and Materials, 1970.

9. ASTM G22, "Determining Resistance of Plastics to Bacteria," American Society for Testing and Materials, 1976.

10. R. Bartha, D. Pramer, "Features of a Flask and Method for Measuring the Persistence and Biological Effects of Pesticides in Soils," Soil Science 100 (1), pp. 68-70, 1965.

11. ANS 16.1, "Measurement of the Leachability of Solidified Low-Level Radioactive Wastes," American Nuclear Society Draft Standard, April 1981.

Appendix A

Cement Stabilization

I. INTRODUCTION

This Appendix to the Technical Position on Waste Form provides guidance to waste generators and processors who intend to use cementitious materials such as Portland and pozzolonic-type cements to solidify and stabilize low-level radioactive wastes in accordance with the requirements of 10 CFR Part 61 (Ref. A1(a)). This guidance is applicable for cementious waste forms destined for disposal in shallow-land disposal sites and engineered structures where the regulatory authorities require stable waste forms. It is expected that the guidance described herein would be used by NRC staff in any Topical Report evaluation of the acceptability of cement waste forms for disposal at near-surface disposal facilities. Waste generators using cement solidification systems and media not approved generically through the Topical Report review process may use this guidance to conduct testing to demonstrate that waste forms satisfy the requirements of Part 61. NRC regulation 10 CFR 20.311 (Ref. A1(b)) requires waste generators to certify that their waste forms meet the requirements of Part 61 (including the requirements for structural stability). Waste generators whose cement waste formulations meet the provisions of this Technical Position will be able to certify that the formulations meet the requirements of Part 61. The disposal site regulatory authorities, however, have the ultimate reponsibility for accepting or rejecting the waste.

Portland and pozzolonic cements have been observed to exhibit unique chemical and physical interactive behavior when used with certain materials and chemicals encountered in some low-level radioactive waste streams. Therefore, this Appendix specifically addresses cement waste form qualification only and is not intended to be applied generically to all stabilization agents (although many of the provisions discussed are, in principle, applicable to other media). This Appendix thus complements, and does not replace, the main body of the Technical Position on Waste Form.

Included in this Appendix are descriptions of methods that may be used in cement waste form qualification testing. Associated acceptance criteria that may be used by NRC staff or others to evaluate the acceptability of the test results are also provided. Included in this waste form testing guidance are descriptions of acceptable procedures for sample preparation and statistical treatment of data. In addition, this Appendix provides guidance on waste stream characterization, process control program (PCP) recipe qualification and specimen examination, surveillance specimen preparation and testing, and procedures for reporting of cement waste form preparation mishaps. This guidance on cement waste forms is intended to provide the best available information on an acceptable approach for demonstrating that a cement-solidified low-level radioactive waste form will possess the long-term (300-year) structural stability that is required by Part 61 for Class B and Class C wastes.

Linkage between the waste form qualification test recommendations in this Technical Position and the requirements of Part 61 is provided in 10 CFR 61.56(b)(1), where it is stated that "a structurally stable waste form will generally maintain its physical dimensions and form, under the expected disposal conditions such as weight of overburden and compaction equipment, the presence of moisture and microbial activity, and internal factors such as radiation effects and chemical changes." The discussion provided in Section II of this Appendix addresses the details of the test procedures and acceptance criteria recommended for cement-stabilized wastes. Further information on test specimen preparation and analysis of data is provided in Section III and Section IV, respectively.

II. WASTE FORM QUALIFICATION TESTING

A. General

As indicated in Section C.2 of the main body of this Technical Position, generic test data may be used "for qualifying process control programs." That is, a low-level radioactive waste generator/processor may perform qualification testing, as described in the following subsections of this Appendix, to qualify recipes for a range of waste compositions (concentrations and loadings) for a given type of waste stream. It is incumbent upon the party providing 10 CFR 20.311 certification, however, to show that the composition(s) of the waste form specimens used in the qualification testing adequately covers the range of waste compositions that will be encountered in the field. An acceptable approach to qualification testing is to perform the tests not only at the maximum waste loading but also at lower loadings (at least one), with appropriate variations in water/cement ratios and proportions of additives. It should not be necessary to perform all the qualification tests for all of the waste loadings, but adequate justifications should be provided for any omissions.

Each individual waste stream should be qualified with test data obtained for that specific waste stream. In cases where two or more waste streams are combined, it should be demonstrated that the specimen compositions used in the qualification testing adequately cover the range of compositions that are intended to be stabilized in the field. This may be accomplished by performing the full series of qualification tests on the "worst-case" composition only, along with one or more tests on alternate compositions, sufficient to show that the selected "worst-case" was chosen correctly.

B. Compression

It is stated in 10 CFR 61.56(b)(1) that "a structurally stable waste form will generally maintain its physical dimensions and form under expected disposal conditions such as weight of overburden and compaction equipment...." Assuming a cover material density of 120 lbs./cu.ft., a minimum compressive strength criterion of 50 psi was established in section C.2.b. of the 1983 Revision 0 portion of this Technical Position. To reflect the increase in burial depth (from 45 to 55 feet) at Hanford, Washington, the minimum compressive strength criterion for generic waste forms was later increased from 50 to 60 psi.

However, as further noted in the above-cited section C.2.a., for solidification agents that are easily capable of meeting the 50 (now 60) psi minimum compressive strength, the waste forms should achieve "maximum practical compressive strengths," not just the "minimum acceptable compressive strength." This provision was included in the Rev. 0, 1983 Technical Position in recognition of the fact that mere resistance to deformation under burial loads is, in itself, inadequate evidence that the waste form microconstituents are bonded together sufficiently well to ensure that the waste form will not over time fall apart due to internal stresses that are chemically, physically, or irradiation induced.

Portland cement mortars, which are comprised of mixtures of cement, lime, silica sand and water, are readily capable of achieving compressive strengths of 5000 to 6000 psi; that is approximately two orders of magnitude greater than the minimum compressive strength required to resist deformation under load in current low-level waste burial trenches. Therefore, to provide greater assurance that there will be sufficient cementitious material present in the waste form to not only withstand the burial loads, but also to maintain general "dimensions and form" (i.e., to not disintegrate) over time, it is recommended that cement-stabilized waste forms possess compressive strengths that are representative of the values that are reasonably achievable with current cement solidification processes. Taking into consideration the fact that low-level radioactive waste material constituents are not in most cases capable of providing the physical and chemical functions of silica sand in a cement mortar, a mean compressive strength equal to or greater than 500 psi is recommended for waste form specimens cured for a minimum of 28 days (see Section III.B of Appendix A). This value of compressive strength is recommended as a practical strength value that is representative of the quality of cementitious material that should be used in the waste form to provide assurance that it will maintain integrity and thus possess the long term structural capability required by Part 61.

Compressive strengths of cement-stabilized waste forms should be determined in accordance with procedures described in ASTM Standard C39: Compressive Strength of Cylindrical Concrete Specimens (Ref. A2). It is recommended that the compressive strength test specimens be right circular cylinders, 2 to 3 inches in diameter, with a length-to-diameter (L/D) ratio of approximately two. Because hydrated cement solids are brittle ceramic materials that fail in tension or shear rather than compression, and at regions of localized stress concentration or microstructural flaw, there tends to be considerable scatter in the strength test data even if all processing variables are kept relatively constant. Therefore, sufficient specimens should be tested to determine the mean compressive strength and standard deviation. Because of the many variables involved, a decision regarding the specific number of specimens to be tested is left to the judgement of the waste processor/qualifier; in no case, however, should the number of as-cured (pre-environmental test) compressive strength test specimens be less than ten. This approach should continue until there are sufficient data available to permit judgements to be made regarding what is reasonably achievable, from a statistical standpoint, in compressive strength testing of low-level waste test specimens. No precision criterion, in the form of an acceptable variance or standard deviation, is recommended at this time.

[For the purposes of verification of Process Control Program (PCP) parameters (see discussion in Section VI of Appendix A), compressive strength tests and/or durometer hardness tests should be performed after the qualification test specimens have been allowed to cure for approximately 24 hours. The results of these tests should be retained and made available for comparison with the results of similar tests that should be performed on PCP specimens fabricated from actual radioactive wastes in the field; (see Appendix A, Section VI.C for details).]

C. Thermal Cycling

Though thermal effects are not called out specifically as an item of concern in 10 CFR 61.56(b)(1), as other factors are, cement-stabilized low-level radioactive waste forms should be demonstrated to be resistant to thermal degradation. There are three basic reasons for this: (1) Section 61.56(b)(1) of Part 61 lists "internal factors" as a condition that must be considered in assuring that a waste form will retain structural stability, and temperature and thermal effects are internal factors; (2) thermal cycling of the waste form will occur, particularly during the storage and transport phase of the waste form's performance "life;" and (3), experience has shown that the thermal cycling test has served well in distinguishing between "strong" and "weak" solidified waste forms. The thermal cycling test imposes a stress (due to differential thermal expansion) between the various microconstituents of the waste form and between different regions of the waste form. By cycling between the maximum and minimum temperatures called for in the test, any cracks initiated in the test specimen may propagate and eventually measurably weaken the waste form. The extent of any degradation that might occur will be a function of various factors such as the amount of cementitious material in the waste form, the bond strength between the materials present, and the morphology of the microconstituents in the waste form microstructure. Thus, the thermal cycling test, by subjecting the waste form specimens to a short-term cyclic thermal stress, challenges the structural capability of the specimens and thus serves as a very useful vehicle for screening out unfavorable "weak" formulations.

The heating and cooling chambers used in determining the thermal cycling resistance of cement-stabilized waste forms should, as stated in Section C.2.b. of the main body of this Technical Position, conform to the description given in ASTM Standard Test Method B553 (Ref. A3). However, because that test method addresses thermal cycling of electroplated plastics, not cement-solidified waste materials, some modifications to the test procedure are necessary. Test specimens suitable for performing compressive strength tests in accordance with ASTM C39 should be used. The specimens should be tested "bare;" i.e., not in a container. Specimens should be placed in the test chamber, and a series of 30 thermal cycles should be carried out in accordance with Section 5.4.1 through 5.4.4 of ASTM B553, with the additional proviso that the specimens should be allowed to come to thermal equilibrium at the high (60 degrees C) and low (-40 degrees C) temperature limits. Thermal equilibrium should be confirmed by measurements of the center temperature of at least one specimen (per test group). A minimum of three specimens for each waste formulation should be subjected to the thermal cycling tests.

Following exposure to 30 thermal cycles the specimens should be examined visually and should be free of any evidence of significant cracking, spalling, or bulk disintegration; i.e., visible evidence of significant degradation would be indicative of failure of the test. Because it is not possible to provide an a priori assessment of the significance of visible defects, taking into consideration the wide range of possible defect configurations, no definition of "significant degradation" is provided here. The organization performing the tests should (1) assess whether visible defects are significant, and (2) obtain and retain photographic evidence of any defects that are judged to be insignificant for future reference. If there are no significant visible defects, the test specimens should be subjected to compression strength testing in accordance with ASTM C39 and should have mean compressive strengths that are equal to or greater than 500 psi.

D. Irradiation

In accordance with the requirements of 10 CFR 61.56(b)(1), and as indicated in Section C.2.c. of the main body of this Technical Position, irradiation testing of solidified waste forms should be conducted on specimens exposed to a minimum dose of 10E+8 rads. The 10E+8 rads radiation dose is approximately equivalent to the dose that would be acquired by a waste form over a 300-year period, if the waste form were loaded to a Cesium-137 or Strontium-90 concentration of 10 Ci/cu.ft. This is the recommended (Ref. A3) maximum activity level for organic resins based on evidence that while a measurable amount of damage to the resin will occur at 10E+8 rads, the amount of damage will have negligible effect on power plant or disposal site safety. However, cementitious materials are not affected by gamma radiation to relatively high cumulative doses (e.g., greater than 10E+9 rads--Ref. A4) considerably in excess of 10E+8 rads. Therefore, for cement-stabilized waste forms, irradiation qualification testing need not be conducted unless (1) the waste forms contain ion exchange resins or other organic media or (2) the expected cumulative dose on waste forms containing other materials is greater than 10E+9 rads. Testing should be performed on specimens exposed to (1) 10E+8 rads or the expected maximum dose greater than 10E+8 rads for waste forms that contain ion exchange resins or other organic media or (2) the expected maximum dose greater than 10E+9 rads for other waste forms. In cases where irradiation testing is warranted, a minimum of three specimens should be tested for each waste formulation being qualified.

Following the irradiation exposure the specimens should be examined visually and should be free of any evidence of significant cracking, spalling, or bulk disintegration; i.e., visible evidence of significant degradation would be indicative of failure of the irradiation test. If there are no significant visible defects (see Section II.C for discussion of "significant degradation"), the test specimens should be subjected to compressive strength testing in accordance with ASTM C39 and should have mean compressive strengths that are equal to or greater than 500 psi.

E. Biodegradation

As indicated in 10 CFR 61.56(b)(1), a structurally stable waste form is one that will be relatively unaffected by "microbial activity." Generic (not specific to type of waste form) recommendations for biodegradation testing provided in Section C.2.e. of the main body of this Technical Position indicate that ASTM Standard Practice G21 (Ref. A5) and G22 (Ref. A6) are suitable methods of test for determining susceptibility to fungi and bacteria, respectively. Experience in biodegradation testing of cement-stabilized waste forms has shown (Refs. A7-A9), however, that they generally do not support fungal or bacterial growth. The principal reason for this appears to be that the fungi and microbes used in the G21 and G22 tests require a source of carbon for growth, and in the absence of any carbonaceous materials in the waste stream, there is no internal food source available for culture growth. Consequently, for cement-stabilized waste forms, biodegradation qualification testing need not be conducted unless the waste forms contain carbonaceous materials (e.g., ion exchange resins or oils).

For cement-stabilized waste forms containing carbonaceous materials, there should be no evidence of culture growth during the G21 and G22 tests. The test specimens (at least three for each organic waste stream formulation being qualified) should also be free of any evidence of significant cracking, spalling or bulk disintegration; i.e., visible evidence of significant degradation would be indicative of failure of the test. If there are no significant visable defects following the test exposures (see Section II.C of this Appendix for discussion of "significant degradation"), the test specimens should be subjected to compression strength testing in accordance with ASTM C39 and should be shown to have mean compressive strengths equal to or greater than 500 psi.

F. Leach Testing

Resistance to leaching of radionuclides is not specifically mentioned in Part 61, nor is radionuclide containment called out as a specific requirement for low-level waste packages. Minimization of contact of waste by water is a fundamental concern of Part 61, however, as evidenced by the statement in Section 61.7 that "...a cornerstone of the system is stability...so that . . access of water to the waste can be minimized (emphasis added). Migration of radionuclides is thus minimized..." In addition, there are several statements in Section 61.51 that address minimization of contact of water with waste. These statements are in recognition of the fact that contact of waste with water is the first step in a potentially major pathway for radionuclide release and migration off-site. Thus, "leaching," or release of radionuclides from a waste form through contact with water is a first step in subsequent migration of the radionuclides from the waste through the groundwater and off the site. Therefore, leaching is a phenomenon that is of fundamental interest in waste disposal.

The leach testing procedure specified in Section C.2.e. of the main body of this Technical Position is ANSI/ANS 16.1: Measurement of the Leachability of Solidified Low-Level Radioactive Wastes by a Short-Term Test Procedure (Ref. A10). In the ANS/ANSI 16.1 test, a test specimen is completely immersed in a measured volume of water, which is changed on a prescribed schedule. Upon removal, the leachant is analyzed for the radionuclides (or elements) of interest. The data obtained by this procedure are expressed as a material parameter of the leachability of each leached species. This parameter is called the "Leachability Index" (L), which is the arithmetic mean of the L values obtained for each leaching interval (where the L value is the logarithm of the inverse of the effective diffusivity). The leachability index, as calculated in accordance with ANSI/ANS 16.1, should be greater than 6.0.

The period of time specified for the leach test in the above-cited Section C.2.e. of this Technical Position is a minimum of 90 days, and the test period called out in the Standard corresponds to 90 days. This time period was selected as a means of determining whether there might be a change in leach mechanism with time; (as explained in the Standard, early leach rates observed with solidified waste forms are most often explained by diffusion--other mechanisms, such as erosion, dissolution, or corrosion, would generally be discernible only after longer leaching times). However, any leaching that involves other mechanisms such as erosion, dissolution, corrosion or other chemical or physical phenomena would most likely be readily observed visually and through mechanical testing. Such observations would be made as part of the immersion test, which is a 90-day test. These facts, coupled with comparisons of 5-day and 90-day data (Ref. A11) on cement waste forms that showed that the percentage differences between 5-day and 90-day leach indices were relatively small for most specimens, indicate that a 5-day leach testing period is sufficient for cement-solidified wastes.

The leachant specified in ANSI/ANS 16.1 is deionized water. It is stated in the above-cited Section C.2.e. of this Technical Position that additional testing using other leachants should also be performed to confirm the solidification agents leach resistance in other leachant media. Synthesized sea water leachant is listed as a preferred alternate leachant. The basis for this is, that while leachability indices are generally lower (i.e., leach rates are higher) for tests conducted in demineralized water than in sea water (Ref. A11), this is not true in all cases for all waste streams. For reasons of economy, however, it is desirable to limit the bulk of the testing to one leachant. If it can be shown that the chosen leachant is the most aggressive one, testing with one leachant is appropriate. Since it is not possible to initially predict (Ref. A9) which leachant (deionized water or synthesized seas water) would be most aggressive, sufficient preliminary testing should be conducted to identify the most aggressive leachant for each waste form formulation being qualified, and that leachant should be used for the balance of the testing (if only one is used). An acceptable method of identifying the most aggressive leachant is to perform 24 hour (or longer) leaching measurements on both leachants and to use the leachant that resulted in the lowest leach indices (i.e., highest leach rate) for the remaining days of testing.

G. Immersion Testing

No "Standard Method of Test" for immersion testing has been adopted for low-level radioactive waste, but as indicated in Section C.2.f. of the main body of this Technical Position, immersion testing may be performed in conjunction with the leach testing (which is to be performed in accordance with ANSI/ANS 16.1). However, in contrast with the period of time (5 days) necessary for leach testing of cement-stabilized wastes, immersion testing should be performed for a minimum period of 90 days. The immersion testing should be performed in either deionized water or synthesized sea water. The immersion liquid should be selected on the basis of short-term (24-hour or longer) leach tests that identify the most aggressive immersion medium (see discussion of leach testing).

The test specimens (at least three for each waste stream formulation being qualified) should be cured for a minimum cure time of 28 days (see Section III, "Specimen Preparation," of Appendix A for details) prior to being immersed. Following immersion, the specimens should be examined visually and should be free of any evidence of significant cracking, spalling, or bulk disintegration. If there are no significant visible defects (see Section II.C of this Appendix for discussion of "significant degradation"), the specimens should be subjected to compressive strength testing in accordance with ASTM C39 and should have post-immersion mean compressive strengths that are equal to or greater than 500 psi and not less than 75 percent of the pre-immersion test (i.e., as-cured) mean compressive strength. If the post-immersion mean compressive strength is less than 75 percent of the as-cured specimens' pre-immersion mean compressive strength, (but not less than 500 psi) the immersion testing interval should be extended (using additional specimens) to a minimum of 180 days. For these cases, sufficient compressive strength testing should be conducted (for example, after 120, 150, and 180 days of immersion) to establish that the compressive strengths level off and do not continue to decline with time.

For certain waste streams (viz., bead resins, chelates, filter sludges, and floor drain wastes) that have been found to exhibit complex relationships of cure time and immersion resistance (Ref. A12), additional immersion testing should be performed on specimens that have been cured (in sealed containers) for a minimum of 180 days. The immersion period should be for a minimum of 7 days, followed by a drying period of 7 days in ambient air at a minimum temperature of 20 degrees Celsius. After the specimens are dried, they should meet the post-immersion test visual and compressive strength criteria specified above.

H. Free Standing Liquids

It is stated in 10 CFR 61.56(b)(2) that "...liquid wastes, or wastes containing liquid, must be converted into a form that contains as little free standing or noncorrosive liquid as is reasonably achievable, but in no case shall the liquid exceed...0.5% of the volume of the waste for waste processed to a stable form." Correspondingly, waste test specimens should have less than 0.5 percent by volume of the waste specimen volume as free liquids as measured using the method described in Appendix 2 of ANSI/ANS 55.1 (Ref. A13). Inasmuch as cement is an alkaline material, evidence of acidic free liquids is indicative of improper waste form preparation or curing. Therefore, any free liquid from cement-stabilized waste forms should have a minimum pH of 9.

I. Full-scale Testing

It is expected that the testing performed in accordance with the guidance provided in Sections A through H above will be carried out on small, laboratory scale specimens. As indicated in Section C.2.h. of the main body of this Technical Position, therefore, it is necessary to correlate the characteristics of full-size products with those of laboratory size specimens. The full-scale specimens should be fabricated using solidification equipment that is the same as or comparable to that used in processing real low-level waste forms in the field. The correlation of full-scale product characteristics should be accomplished by performing (1) compressive strength tests on as-cured material (cured for a minimum of 28 days), and (2) 90-day immersion tests that include post-immersion compressive strength tests (See Section II.G above) for the most conservative waste stream(s) being qualified.

Test specimens obtained from the full-scale waste forms by coring or sectioning should be destructively analyzed to ensure that the product produced is homogeneous to the extent that all regions in the product can expect to have compressive strengths that meet the criteria called out in Section II.B above.

III. QUALIFICATION TEST SPECIMEN PREPARATION

A. Mixing

Experience in preparation of lab-scale and full-scale cement-solidified waste forms (Ref. A9) has shown that the method employed in mixing the ingredients can have a dramatic influence on the reactivity of the materials, the structure of the solidified waste form, and the resultant properties and characteristics of the waste form. Important parameters include type of equipment and mixing time because they will determine the amount of energy imparted to the ingredients used in the solidification recipe. This is especially important in cases where properties and characteristics of small, lab-scale specimens are used to predict the behavior of large, full-scale products. In preparing laboratory-sized qualification test specimens, it should be shown by analysis and/or testing that the type of equipment used, the mixing time, the speed of the mixer, etc. will, in combination, impart the same degree of mixing to the laboratory specimens as the full-scale mixing equipment and procedure will impart to full-scale waste forms and that the degree of mixing is sufficient to ensure production of homogeneous waste forms.

B. Curing

The curing conditions for small, laboratory-scale qualification test specimens, should, to the extent practical, be the same as the conditions obtained with full-scale products. Inasmuch as cement constituents exhibit a significant exothermic heat of hydration, while possessing low thermal conductivity, the interior temperature of large, full-scale cement waste forms may be elevated significantly (approaching even the boiling point of water). To ensure that the laboratory specimens endure curing conditions that are reasonably similar to those of full-size products, the waste form centerline temperature profile as a function of time should be obtained for the largest full-sized waste form to be qualified for each waste stream. That profile should be duplicated, to the extent practical, in the laboratory specimens. An acceptable method is to cure the specimens in a suitable oven for a period of time equivalent to the peak heat of hydration period. For the purposes of this Technical Position that period of time is taken to be that required for the centerline temperature of a full-scale waste form to decrease to a near-ambient (30 degrees Celsius or lower) temperature level

Care should be taken to ensure that the waste loadings and cement concentrations in the full-scale waste forms provide sufficient margin to preclude reaching the boiling point of the pre-solidification mix. This is necessary to ensure that the waste form formulations will not be subject to uncontrolled variations due to water losses caused by evaporation during set. Uncontrolled porosities due to vapor bubble formation and rapid set due to elevated temperatures will also be avoided by limiting the maximum temperatures in the cement-solidified waste forms.

The compressive strength of hydrated cement and concrete solids increases asymptotically as the mixtures cure. Normally, the strength at 28 days approaches seventy-five percent or more of the "peak" value, though when pozzolonic cements are used the time required to reach peak strength may be extended. Sufficient test specimens should be prepared to determine the compressive strength increase with time to ensure that the specimens have attained sufficient (i.e., greater than 75% of the projected peak) strength prior to subjecting the remaining specimens to the qualification testing called out in Sections II.C through II.G. of this Appendix.

C. Storage

Test specimens that will be subjected to the qualification testing described in Section II of this Appendix should be kept in sealed containers during curing and storage. This is intended to simulate the environment that would be obtained in a typical full-scale waste form liner and will prevent loss of water that might affect the performance of the waste form specimens during subsequent testing.

IV. STATISTICAL SAMPLING AND ANALYSIS

As noted in the discussion of compressive strength testing (see Section II.B above), there tends to be considerable scatter in the compressive strength data obtained on brittle ceramic materials such as cement. Therefore, sufficient specimens should be tested in the as-cured condition to provide enough data to establish a mean and standard deviation, though for reasons discussed in Appendix A Section II.B, the number of as-cured specimens to be tested is left to the judgement of the waste formulation qualifier. For statistical purposes, however, the number of as-cured (pre-environmental test) compressive strength specimens should be ten or greater for a given formulation. Further discussion of the rationale for this provision is provided in Section II.B of this Appendix. For the minimum quantities of test specimens recommended in the respective subsections of this Appendix, the specimens tested should have a post-test mean compressive strength that is equal to or greater than 500 psi. Note that for the immersion tests, a slightly different acceptance criterion is identified, in subsection II.G of this Appendix. Variations in individual specimen compression strength need not be considered.

Other than the determinations of compressive strength, the only other parameter of interest in qualification testing of low-level waste forms that lends itself to statistical treatment is the leachability index. ANSI/ANS 16.1 (Ref. A10) uses the confidence range and correlation coefficient as measures of discrepancies in the measurements of leachability. The Standard requires that the confidence range and correlation coefficient be reported with the Leachability Index. As is the case of the ASTM C39 Compressive Strength standard, however, no precision criterion has been established yet for the ANSI/ANS 16.1 leach test.

V. WASTE CHARACTERIZATION

The importance of waste characterization was extensively discussed at the May/June Workshop on Cement Stabilization of Low-Level Radioactive Waste that was held in Gaithersburg, MD. The Proceedings (Ref. A9) of the Workshop, particularly the efforts of Working Group 4, record the discussions and provide useful information on the routine characterization of typical waste streams. Waste characterization would typically be expected to include as a minimum the identification of major constituents in the waste (including primary ions and salts or other solids), density, pH, temperature, radioactive isotopes, and a check for the presence of secondary ingredients that could significantly affect the hydration of the cement.

Some waste streams, such as pressurized water reactor (PWR) primary coolant system borated water, are relatively well-characterized and free of secondary ingredients. There are other waste streams, however, such as ion exchange resins, filter sludges and floor drain liquids, that may contain chemicals that can significantly retard or accelerate the hydration of cement or in other ways adversely affect cement waste form performance (Ref. A9). It is impractical for a waste processor to perform qualification testing on every possible combination and concentration of secondary constituents in a given type of waste stream. Nor is it considered practical or necessary for a waste generator to perform a complete quantitative chemical analysis on every batch of waste that is produced. It is, however, incumbent on radwaste system managers and processors to be cognizant of the types of chemicals that may produce problems in using cement in the solidification and stabilization of low-level radioactive waste. The introduction of such chemicals into waste treatment systems that utilize cement stabilization media should be avoided or specifically compensated for in the formula used for stabilizing that waste stream. If the waste processor is a vendor or is otherwise not the generator of the waste, it is incumbent on all parties to be in adequate communication with each other with regard to the types and quantities of chemical ingredients in the waste and the capability of the waste formulation to provide long-term structural stability to the waste form. As a part of process control, mixing of different wastes in holding tanks and transfer of liquid wastes without adequate flushing of lines should be generally avoided, because such mixing might introduce ingredients into the waste that were not present in the qualification test program that was conducted for the waste stream in question.

To assist waste generators and processors in developing a sense of greater awareness of low-level radioactive waste stream ingredients that may adversely affect the setting and stability of cement-solidified waste forms, a list of such chemicals is provided in Table I. This list is not intended to be all-inclusive. Moreover, some of the constituents listed may be considered hazardous materials, as defined by Environmental Protection Agency (EPA) criteria, and which thus, if mixed with radioactive material, could be classified as a "mixed waste." Any questions about low-level radioactive wastes that might be classified as mixed wastes should be directed to the EPA.

Low-level radioactive waste generators and processors who intend to stabilize Class B and Class C waste with cement should either (a) prevent the contamination of, (b) limit to the extent practical, or (c) pre-treat as appropriate, waste streams that may contain the chemicals and constituents in Table I. It is the responsibility of the waste generator and processor to ensure that the cement formulation used for a given waste stream is qualified for the waste stream chemical constituents and concentrations in question.

VI. PCP SPECIMEN PREPARATION AND EXAMINATION

A. General

The purpose of a Process Control Program (PCP) is to describe the envelope within which processing and packaging of low-level radioactive wastes will be accomplished to provide reasonable assurance of compliance with low-level waste requirements. All commercial nuclear power plants have plant specific PCPs. The guidance provided in this section of this Appendix is not, however, intended to address facility-specific PCPs, which, in addition to containing a general description of the methods for controlling the processing and packaging of radioactive waste, may also contain a description of the system and operating procedures, instructions on manifest preparation, and a discussion of administrative controls. Rather, this guidance addresses only the recipe portion of cement stabilization of low-level waste; that is, the guidance addresses the nature of the information that should be provided in a generic PCP concerning the type and quantity of ingredients used in the cement waste form formulation, the order of addition, and the method, process, and time required for mixing the ingredients in the preparation of verification and surveillance specimens as well as the full-scale waste forms. Also provided is guidance on the preparation of PCP "verification" and surveillance specimens and the type of examinations and testing that should be performed on those specimens.

This information on verification specimens is intended to provide assurance that the formulations used in the qualification testing program correspond to those actually used in the field. The surveillance specimen program, described in Section VII of this Appendix, is intended to provide verification that the waste forms are remaining stable with time.

For each low-level radioactive waste formulation, the generic PCP should address the boundary conditions (i.e., bounding process parameters) for processing the waste to provide reasonable assurance that the final waste form will meet 10 CFR Part 61 stability requirements. The process parameters will be influenced by (a) the characteristics of the waste prior to processing, (b) the qualities of the solidification medium, as influenced by additives, and (c) the physical/chemical process of preparing the waste into a final waste form. Variables that influence the process and have an effect on the product, and that should be, therefore, be identified and restricted within acceptable bounds for each waste form include the following:

1. Type of waste (e.g., bead resin, including type--anion/cation/mixed/ manufacturer/weak acid/strong acid, percent depleted, powdered resins, boric acid, sludges);

2. Waste characteristics having influence on the final waste form (e.g., pH, oil content, chelating agents, water content, maximum concentration of secondary ingredients);

3. Additives (e.g., type of cement, water, lime, silica fume, fly ash, furnace slag,) and the order of addition;

4. Physical process parameters (e.g., maximum temperature, mixing equipment required, mixing and curing times).

The generic PCP should indicate how representative samples of the feed waste are to be obtained for preparing PCP verification and surveillance specimens. The PCP should identify typical and maximum batch sizes and the number of PCP specimens to be taken for each batch. The PCP should describe where adjustments could be made to the feed waste material, in the event that certain feed material parameters that may be encountered in the field fall outside of the acceptable range for processing. These adjustments should not be undertaken if the resultant waste stream feed material and stabilized waste form were to be chemically or physically different from that qualified in laboratory testing.

If, during the course of full-scale waste form preparation at a nuclear power plant, it should become necessary to effect an <u>ad hoc, impromptu</u> change in the approved recipe or procedure to avoid an incomplete or otherwise unsatisfactory solidification condition, the change should be reviewed and approved by the facility licensee pursuant to the provisions of 10 CFR 50.59. This process should be followed in all such cases where ad hoc changes are necessary whether or not a generic PCP has received approval as part of a Topical Report review process. Inasmuch as the affected waste form would lack assurance of long-term structural stability (because it was produced under conditions that were outside of the envelope of the conditions used in the qualification tests), it is anticipated that the resultant waste form would not be accepted for disposal at a disposal site without the expressed approval of the disposal site regulatory authorities. It is also anticipated that, prior to accepting the waste, the regulatory authority would require either (1) adequate mitigation of any potential adverse effects on the long-term structural stability of the waste form or (2) an acceptable justification concerning the lack of any potential significant effect of the affected waste form on the overall performance of the facility. Alternatively, the disposal site regulatory authority could accept the affected waste for disposal with the provision that the required structural stability would be provided at the disposal facility by means of an engineered structure.

After the generic PCP has been reviewed and approved by the NRC, the PCP parameters and procedures should be followed as described in the Topical Report (or other documentation) so that the 10 CFR 20.311 certification can be made without the need for additional justification that the cement-solidified waste meets the requirements of 10 CFR Part 61. Once a generic PCP has been approved by the NRC any subsequent changes to the generic PCP should be reviewed and approved by the NRC. Any incomplete or otherwise unsatisfactory solidification condition known to waste generators and processors is requested to be reported to the NRC (Director, Division of Low-Level Waste Management and Decommissioning) within 30 days after such an occurrence is known (see Section VIII). The actions taken to produce an acceptable waste form after the initial unsatisfactory solidification condition was identified should be described.

B. Preparation of PCP Specimens

Prior to plant-specific solidification of full-scale waste forms, representative samples of the feed waste should be obtained in sufficient quantity to prepare the desired number of PCP specimens. The feed waste material should be solidified using the recipe that has been qualified in laboratory testing for the given waste stream. Mixing of the waste materials with the cement and additives should be accomplished in a manner that duplicates, to the extent practical, the mixing conditions that are obtained with full-scale mixing. The specimens should be cured under conditions similar to those used in the laboratory qualification test program. PCP specimens should be prepared for each batch of waste that is required to meet the 10 CFR Part 61 structural stability criteria. For the purposes of the guidance provided in this Technical Position, a "batch" is herein defined as any quantity of waste stream feed material that is from a single source (e.g., a holding tank), that is processed as a single batch (even though it maybe subdivided in more than one unit waste form; e.g., liner), and that, therefore, possesses unvaried, single operation, batch characteristics.

C. PCP Specimen Examinations and Testing

1. Short-term (24-hour PCP Verification) Specimens -

Prior to solidifying full-scale waste forms, plant-specific PCP verification specimens should be prepared, in accordance with procedures described above, for examination and compressive strength testing. The specimens should be free of significant visible defects, such as cracking, spalling or disintegration and should exhibit less than 0.5% by volume of the specimen as free liquid. As a measure of process control, the specimens should, within a 24-hour period after preparation, be subjected to an ASTM C39 compressive strength test; (penetrometer measurements may be substituted, as described below). The compressive strength values should be within two standard deviations of the mean compressive strength values obtained at 24 hours for test specimens prepared and tested as part of the associated laboratory generic qualification test program for the waste formulation. Alternatively, penetrometer tests can be used in lieu of C39 compressive strength measurements if acceptable correlation data demonstrating the relationship between the compressive strength values and penetrometer values have been obtained for the waste stream formulation in question. If penetrometer tests are used, the mean penetrometer hardness values obtained on the verification specimens should be within two standard deviations of the mean obtained on the qualification test specimens for that formulation. If the compressive strength or penetrometer measurements do not meet the above criteria, a second set of PCP specimens should be prepared and retested. The second set of PCP specimens should be fabricated using either the same formula or an adjusted one that falls within the compositional envelope of the qualification tests conducted for that waste stream.

2. Long-term Surveillance Specimens -

The guidance herein addressing long-term surveillance specimens is directly applicable to waste generators and to vendors processing wastes at licensed facilities who intend to certify, in accordance with the provisions of 10 CFR 20.311, that the cement-solidified waste meets the structural stability requirements of 10 CFR Part ii. Sufficient PCP specimens should be prepared to permit the retention, examination and testing of surveillance specimens. The surveillance specimens should be stored in sealed containers at normal room temperatures. The examination and testing of surveillance specimens is described in Section VII of this Appendix.

VII. SURVEILLANCE SPECIMENS

The purpose of the surveillance specimens is to provide confirmation that the waste forms prepared for certain waste streams, (in particular bead resins, chelates, filter sludges, and floor drain wastes) are performing as expected. At periods of time equal to 6 months and 12 months after preparation, the surveillance specimens should be examined visually and should be free of evidence of significant cracking, spalling or bulk disintegration (see Section II.C of Appendix A for discussion of "significant degradation"). At least one specimen should be subjected to an ASTM C39 compressive strength (or penetrometer) test at the 6 and 12 month periods. The mean compression strength (or penetrometer) value(s) obtained should be not more than two standard deviations below the mean of the as-cured strength or penetrometer values obtained with the qualification test specimens cured for an equivalent period of time.

At 12 months after preparation, one or more PCP surveillance specimens should be subjected to an immersion test. The duration of the immersion test should be a minimum of 14 days. Upon removal from the immersion liquid, which should be either deionized water or synthesized sea water (see Section II.F of this Appendix) the specimens should be allowed to dry in ambient air for a minimum of 48 hours. The specimens should then be examined visually and should be free of significant surface or bulk defects such as cracking, spalling, or bulk disintegration. Following the immersion test, the specimen(s) should be subjected to an ASTM C39 compressive strength (or penetrometer) test. The test results should meet the criteria discussed above.

If the PCP surveillance specimens tested either by the vendor of an NRC-approved Topical Report or by a utility or other licensee, should fail any of the above tests, the wastes previously solidified may not meet the stability requirements of 10 CFR Part 61. Therefore, the NRC (Director, Division of Waste Management and Decommissioning) and licensee (if other than the waste processor that shipped the suspect waste to the disposal facility) should be notified in writing within 30 days. In turn, the licensee should notify the disposal facility operator and regulatory authority if the 10 CFR 20.311 certification as to waste stability was invalidated by this finding. The licensee's report should satisfy the information needs of the regulatory authority and should describe the waste stream solidified, the waste formulation used, the number of full-scale waste forms that had been produced, date of shipment, manifest numbers, and the results of the tests. The report should also contain a discussion of the significance of the test results and proposed changes, if any, that might have to be made to the waste formulation to ensure that, for the waste stream in question, future waste forms would be stable.

For all waste processors (including utility licensees and vendors of NRC-approved Topical Reports), it is recommended that a summary report that addresses the results of PCP surveillance specimen preparations and examinations should be prepared annually by the waste processor and submitted to the NRC (Director, Division of Waste Management and Decommissioning). The report should document the results of all visual examinations and immersion, compression, and/or penetometer tests performed on the cement-stabilized waste form surveillance specimens during the calendar year. The annual report should be submitted within 90 days of the end of each calendar year. A commitment to provide this information will be made a condition of approval for all future license applications, topical report submittals or other regulatory actions that deal with cement waste forms, where the waste generators and/or processors desire NRC endorsement of their 10 CFR 20.311 certifications.

VIII. REPORTING OF MISHAPS

Known cement waste form processing mishaps, including but not restricted to, cement waste forms that have not solidified completely, waste forms that have swelled and/or disintegrated, waste forms that were not prepared in accordance with an approved PCP, and waste form preparations that resulted in unusual exothermic reactions, should be reported by the cognizant waste processor to the NRC (Director of the Division of Waste Management and Decommissioning) within 30 days of the time that the vendor becomes aware of the incident. Licensees should also report such mishaps to the disposal site regulatory authority since such an event may indicate the waste form will or does not satisfy the stability requirements of 10 CFR Part 61. If the mishap becomes known to the waste generator and/or processor before the waste forms are shipped off-site, the affected waste form(s) should not be shipped until approval is obtained from the disposal site regulatory authority. A commitment to report and deal with waste form mishaps as discussed above will be made a condition of approval for all future license applications, topical report submittals, or other regulatory actions that deal with cement waste forms, where the waste generators and/or processors desire NRC endorsement of their 10 CFR 20.311 certifications.

IX. IMPLEMENTATION

This Appendix to the Technical Position on Waste Form reflects the current NRC staff position on an acceptable means for meeting the 10 CFR Part 61 structural stability requirements for cement waste forms. Therefore, except in those cases in which the waste generator, vendor, and/or processor proposes an acceptable alternative method for complying with the stability requirements of 10 CFR Part 61, the guidance described herein will be used by the NRC staff in all future evaluations of the acceptability of cement waste forms for disposal at near-surface disposal facilities.

X. REFERENCES

A1(a). Part 61 - Licensing Requirements for Land Disposal of Radioactive Waste, Code of Federal Regulations, Title 10: Energy.

A1(b). "Method for Obtaining Approval of Proposed Disposal Procedures," Subsection 311 of Part 20 (20.302), Code of Federal Regulations, Title 10: Energy.

A2. American Society for Testing and Materials Standard Test Method for Compressive Strength of Cylindrical Concrete Specimens, ASTM C39, October 1984.

A3. D.R. MacKenzie, M. Lin, and R.E. Barletta, "Permissible Radionuclide Loading for Organic Ion Exchange Resins for Nuclear Power Plants," Brookhaven National Laboratory Draft Report, BNL-NUREG-30668, January 1982.

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A5. American Society for Testing and Materials Standard Practice for Determining Resistance of Synthetic Polymeric Materials to Fungi, ASTM G21, 1985.

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A9. Proceedings of the Workshop on Cement Stabilization of Low-Level Radioactive Waste, U.S. Regulatory Commission Report, NUREG/CP-0130, (in preparation).

A10. American National Standards Institute/American Nuclear Society American National Standard for Measurement of the Leachability of Solidified Low-Level Radioactive Wastes by a Short-Term Test Procedure, ANSI/ANS 16.1-1986, April 14, 1986.

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A12. P. L. Piciulo, J. W. Adams, J. H. Clinton, and B. Siskind, "The Effect of Cure Conditions on the Stability of Cement waste Forms after Immersion in Water," Brookhaven National Laboratory Report, WM-3171-4, August 1987.

A13. American National Standards Institute/American Nuclear Society American National Standard for Solid Radioactive Waste Processing System for Light Water Cooled Reactor Plants, Appendix 2, March 1979.

Table I

LIST OF WASTE CONSTITUENTS THAT MAY CAUSE PROBLEMS WITH CEMENT SOLIDIFICATION

POTENTIAL PROBLEM CONSTITUENTS WHICH MAY BE EXPECTED IN THE WASTE STREAM

Inorganic Constituents

- Borates [1]
- Phosphates [1]
- Lead salts [2]
- Zinc salts
- Ammonia and ammonium salts
- Ferric salts
- "Oxidizing agents" [1] (often proprietary)
  - Permanganates [1]
  - Chromates [2]
- Nitrates [1]
- Sulfates [1]

Organic Constituents - Aqueous Solutions

- Organic acids [1]
  - Formic acid (and formates)
- "Chelates" [1],[3]
  - Oxalic acid (and oxalates)
  - Citric acid (and citrates)
  - Picolinic acid (and picolinates)
  - EDTA (and its salts)
  - NTA (and its salts)
- "Decon solutions"[1]
  - Soaps and detergents [1]

Organic Constituents - Oily Wastes

- Benzene [1],[2]
- Toluene [1],[2]
- Hexane [1]
- Miscellaneous hydrocarbons
- Vegetable oil additives

POTENTIAL PROBLEM CONSTITUENTS THAT MAY BE AVOIDED BY HOUSEKEEPING OR PRETREATMENT [4]

Generic Problem Constituents

- Oil [1] and grease
- "Aromatic oils" [1]
- "Organic solvents" [1],[2]
  - Dry-cleaning solvents [1],[2]
- "Industrial cleaners" [1],[2]
  - Paint thinners [1],[2]
- "Decon solutions" [1]
  - Soaps and detergents [1]

Specific Problem Constituents - Organic [5]

- Acetone [1],[2]
- Methyl ethyl ketone [2]
- Trichloroethane [2]
- Trichlorotrifluoroethane [2]
- Xylene [2]
- Dichlorobenzene [2]

Specific Problem Constituents - Inorganic

- Sodium hypochlorite [1]

NOTES:

[1] These constituents have been specifically identified by vendors as having the potential to cause problems with cement solidification of low-level wastes.

[2] The presence of these constituents may result in the generation of mixed wastes. The Environmental Protection Agency should be contacted for more information.

[3] All of these chelating agents could also be identified as "organic acids."

[4] Good housekeeping and pretreatment could also be effective in preventing problems with cement solidification for many of the constituents listed in the top list.

[5] These specific constituents also fall into several of the "generic" problem constituents "categories" listed at the left.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555

September 6, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: REVISION 1 OF DRAFT TECHNICAL POSITION ON WASTE FORM

During its 23rd meeting on August 29 and 30, 1990, the Advisory Committee on Nuclear Waste (ACNW) reviewed a draft version of Revision 1 of the Technical Position on Waste Form, prepared by NRC's Division of Low-Level Waste Management and Decommissioning. The Committee also had the benefit of discussion with the NRC staff on this matter.

The revision represents a significant expansion of the previous document on this same subject and reflects many of the points that were called to the attention of the NRC staff during previous ACNW and ACRS subcommittee meetings. Owing to the importance to public health and safety that is now properly attached to the quality of the low-level waste form, we conclude that this technical position, when fully implemented, can serve as a useful guide in the evaluation of waste forms used in low-level waste disposal. We believe that the required reporting of mishaps will be especially useful.

Listed below are several concerns that the Committee has on this subject. However, we believe that publication of the Technical Position need not be held up pending resolution of these concerns. To assist in their resolution, we recommend that the NRC staff consider the detailed discussions held during the ACNW meeting of August 29, 1990.

1. The applicable regulation (10 CFR Part 61) places emphasis on the physical stability of the waste form (Class B and Class C) with the intent that by this means access of water to the waste can be controlled. There is no requirement in Part 61 for a specified resistance of the waste form to leaching of radionuclides by ground water. We believe that an important attribute of the waste form is its behavior related to migration of radionuclides into the environment. We believe a revision of Part 61 addressing this point is needed, but

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Enclosure 2

until that is completed, the Technical Position should be amended to reflect more directly the attention that leaching resistance should be given. The almost exclusive focus of the Technical Position on mechanical integrity of the waste form and the effect of various phenomena (e.g., thermal cycling, radiation, and immersion in water) on that integrity should be supplemented by requirements that leach resistance, as measured by a specified separate test, should be maintained in parallel with mechanical strength after the waste is subjected to these phenomena.

2. The testing requirements cited in the revised Technical Position should be representative of conditions likely to be encountered in a shallow land burial site. The primary mobilizing agent is ground water which could be more aggressive in enhancing movement of radionuclides than the distilled water or synthetic sea water now specified in the Technical Position. We believe that the specific test conditions cited in the Technical Position, now oriented only to structural impact, should be complemented by additional conditions that relate to the ground water chemistry of the waste. Further, biodegradation tests should be specified for cementitious waste matrices using bacteria that are likely to affect cement as well as the organic component of the waste.

3. We believe that the provisions for tests of the radiation resistance of waste forms may not be sufficiently conservative when considering the potential for hydrogen generation in closed spaces. The NRC staff is urged to reexamine this topic to ensure that slow buildup of hydrogen from water-bearing wastes in sealed containers does not become a problem for long-term, safe disposal.

4. We believe that insufficient attention has been given to the testing of aged waste forms. Many of the matrices, including concrete, that are used to contain wastes continue to change chemically and physically long after their preparation. Owing to the longer term focus (i.e., 300 years) of the waste integrity requirement, definition of the behavior of waste specimens that simulate aged waste forms appears appropriate for inclusion in the Technical Position where such testing appears feasible and reasonably reliable.

5. The Committee notes that a part of the regulatory control over low-level waste disposal is based on Part 20 regulations (10 CFR 20.311). We urge that the NRC staff examine the revisions in Part 20 that affect low-level waste and ensure that the Technical Position and the updated Part 20 are compatible.

6. The Committee is aware that the newly developed criteria for compressive strength of acceptable cementitious waste forms

[500 psi] lacks strong technical justification but was selected to preclude the use of unstable waste forms. The NRC staff should include in the Technical Position recognition that the compressive strength that is initially called for may not be retained by the waste form for its required life. Long-term degradation of compressive strength to lower levels, but not less than the approximately 60 psi required for other waste forms, may be acceptable.

We hope you will find these comments useful.

Sincerely,

Dade W. Moeller

Dade W. Moeller
Chairman

Reference:

U.S. Nuclear Regulatory Commission Draft Technical Position on Waste Form (Revision 1) dated June 1990, Prepared by Technical Branch, Division of Low-Level Waste Management and Decommissioning (Predecisional)

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

June 18, 1990

MEMORANDUM FOR: Richard L. Bangart, Director
Division of Low-Level Waste Management
and Decommissioning, NMSS

FROM: Stuart A. Treby, Assistant General Counsel
for Rulemaking & Fuel Cycle
Office of the General Counsel

SUBJECT: REVISION TO TECHNICAL POSITION ON WASTE FORM

As requested in your memorandum, subject as above, dated May 23, 1990, this office has reviewed the draft revision of the Technical Position (TP) on Waste Form. We have two main areas of concern with the TP, i.e., the information collection requirements contained in the TP and the intent expressed in the TP to place requirements on vendors who are non-licensees, particularly the requirement to maintain radioactive waste for "surveillance" purposes.

Appendix A of the TP contains several recordkeeping and reporting requirements (page A-18). Although the recent Supreme Court case of Dole v. United Steel Workers, No. 88-1434, ___ U.S. ___, Feb 21, 1990, holds that third party notification requirements for safety purposes are not subject to OMB approval, OMB has not yet issued implementing instructions on how agencies should treat such requirements. Aside from that consideration, there are other reporting requirements found on page A-18, which will require OMB clearance under the Paperwork Reduction Act.

The more critical issue raised by the revision is whether the NRC can place any requirements on vendors as non-licensees. Section 161c, in pertinent part, gives the Commission general authority to "make such studies..., obtain such information...as the Commission may deem necessary or proper to assist it in exercising any authority provided in this Act, or in the administration...of this Act, or any regulations...issued thereunder." This provision of the AEA was originally contained in the 1946 Atomic Energy Act and was incorporated verbatim into the 1954 Act. There is almost no legislative history (and that is found only in the legislative history for the 1946 Act) as to Congress' intent in including the provision, other than to reiterate that 161c grants to the Commission general authority to enable it to discharge its responsibilities. See S Rep No. 1211, 79th Cong., 2d Sess., page 27,28 (1946) and HR Rep 2478, 79th Cong., 2d Sess., page 13 (1946). Therefore, in our opinion, the language of this provision can be read in accordance with its common meaning and usage.

As you know, 10 CFR Part 61 was issued under authority of the Atomic Energy Act of 1954, as amended. The revised TP serves to provide additional guidance as to appropriate waste forms which meet the requirements of Part 61.

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Enclosure 3

Accordingly, we believe that there is a legal basis, pursuant to §161c, to seek the information intended to be collected or provided under Appendix A of the TP from a non-licensee, i.e., a vendor(s) (subject to the impact of the Dole case cited above).

On the other hand, we do have difficulty with the apparent requirement for vendors to maintain "Surveillance Specimens" as specified under Section VII, Appendix A, of the TP. While it is not legally objectionable to enter into a quasi-contractual relationship with a vendor for the purpose of providing Topical Report reviews and certification as to a waste form(s) in return for the vendor subsequently providing the information and notifications set out in Appendix A, it is another matter to require the vendor to possess and test radioactive material in the form of a "surveillance specimen." The NRC does not normally allow a "person" (as defined in §11s, AEA) to possess radioactive material, except under a license issued by the Commission. Therefore, it would appear that the impact of the TP is to require the vendor to become a "licensee," at least for the purpose of possessing "surveillance specimens." We suspect that such a condition could chill the submission of Topical Reports in this area. We would have less concern if the TP were more flexible in this regard, for example, to allow the vendor, at its option, to arrange for storage and testing of "specimens" by a licensee (either waste generator or third party) so that the vendor's obligation "under the contract" could be limited to reporting.

Should you have questions concerning this response, please contact Ron Smith, X21640, or Bob Fonner, X21643, of my staff.

Robert L. Fonner
for Stuart A. Treby
Assistant General Coursel
for Rulemaking & Fuel Cycle
Office of the General Counsel

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

October 15, 1988

MEMORANDUM FOR: Victor Stello, Jr.
Executive Director for Operations

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

SUBJECT: MINUTES OF CRGR MEETING NUMBER 147

The Committee to Review Generic Requirements (CRGR) met on Wednesday, September 22, 1988 from 9-12 a.m. A list of attendees for this meeting is attached (Enclosure 1). The following items were addressed at the meeting:

1. B. Sheron (RES) and F. Eltawila (RES) presented for CRGR review staff evaluations of the IDCOR proposed methodologies for performing the Individual Plant Examinations (IPEs) called for in the Commission's Severe Accident Policy Statement. The Committee recommended in favor of issuing the SERs, subject to several clarifications and modifications to be coordinated with the CRGR staff. This matter is discussed in Enclosure 2.

2. G. Bagchi (NRR) and L. Reiter (NRR) briefed the Committee on the staff's review of Topical Report EPRI NP-4726, "Seismic Hazard Methodology for the Central and Eastern United States." This was done by industry, primarily to address concerns raised by USGS that there may be a low probability occurrence of a large earthquake along the eastern seaboard of the U.S. The staff found the methodology to be acceptable for computing probabilistic seismic hazard. The staff indicated that this document does not represent any regulatory action and the staff committed to providing their position regarding regulatory requirements by next spring. The CRGR requested a further briefing on this issue at the appropriate time. A copy of the briefing slides used by the staff at this meeting are included as Enclosure 3.

3. J. Greeves (NMSS) and J. Surmeier (NMSS) briefed the Committee on the status of NMSS waste form activities. The staff discussed the status of the implementation of Part 61 requirements for waste form. The staff discussed the process and status of topical report reviews on waste forms. The Committee requested to be kept informed regarding the status of the low-level waste topical report reviews, and agreed that CRGR did not have to routinely review staff actions in this area. A copy of the briefing slides used by the staff at this meeting are included as Enclosure 4.

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 the 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.

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Enclosure 4

Questions concerning these meeting minutes should be referred to Cheryl Sakenas (492-4148).

E L Jordan

Edward L. Jordan, Chairman
Committee to Review Generic
Requirements

Enclosures:
As stated

cc w/enclosures:
Commission (5)
SECY
Office Directors
Regional Administrators
CRGR Members
W. Parler
B. Sheron
F. Eltawila
G. Bagchi
L. Reiter
J. Greeves
J. Surmeier
E. Rossi
C. Berlinger

UPDATE CRGR ON NMSS WASTE FORM ACTIVITIES
SEPTEMBER 22, 1988

I. Background
   A. Pre 10 CFR Part 61 Experience -- Sheffield, West Valley, Maxey Flats
   B. The Role of Agreement States in Waste Form Activities
      1. Existing SLB Sites -- Agreement States
      2. Future - Mostly in Agreement States using Engineered Alternatives

II. Part 61 Requirements for Waste Form
   A. Performance Objectives (Subpart C)
   B. Stability of the disposal site after closure (61.44)
      1. Class B & C wastes must have structural stability; generally maintain its physical dimensions and form for 300 years (61.7)
      2. Stability intended to ensure that waste does not
         a. structurally degrade, and
         b. affect overall stability of the site through
            -- slumping
            -- collapse, or
            -- other failure of the disposal unit, and thereby lead to water infiltration
      3. 4 ways to achieve it (waste form, processing, container, or structure)

III. 1983 Branch Technical Position
   A. Provides guidance on how to obtain reasonable assurance of structural stability
   B. Establishes types of tests and acceptance criteria
   C. Provides specificity that Part 61 lacks
   D. Implementation adjusted in several areas since publication
      1. 60 psi versus 50 psi
      2. cement waste form and polyethylene HIC issues

IV. Topical Report Review Process
   A. Agreement States have regulatory authority for LLW disposal
   B. States lack adequate staff to perform technical review
   C. NMSS provides a service by performing "Central" review of TRs
   D. Agreement States may impose more stringent requirements (e.g., stabilized Class A waste forms)

V. Topical Review Status (September 20, 1988 Table)
   A. Approved
   B. Discontinued
   C. Withdrawn
   D. Under Review
      1. Polyehtylene HICs
      2. Cement waste forms
   E. DOE's West Valley Demo Project

VI. The Future
   A. ACNW Interest (September 16, 1988 Letter)
   B. Poly Determination
   C. Cement Determinations
   D. Grandfathering

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

TOPICAL REPORT REVIEW STATUS SUMMARY

SOLIDIFIED WASTE FORM and HIGH INTEGRITY CONTAINERS (HICs)

September 20, 1988

Vendor	Docket No.	Type	Disposition
			Closed
Waste Chem	WM-90***	Solidification (bitumen)	Approved. 1/22/88
General Electric	WM-88	Solidification (polymer)	Approved. 12/27/85
U.S. Gypsum	WM-51***	Solidification (gypsum)*	Approved. 3/3/88
Chichibu	WM-81	HIC (poly impreg/concrete)	Approved. 6/25/86
Nuclear Packaging	WM-45	HIC (ferralium/FL-50)	Approved. 11/7/85
Nuclear Packaging	WM-85***	HIC (ferralium/family)	Approved. 4/20/88
DOW	WM-82***	Solidification (polymer)**	Approved. 6/1/88
ATI	WM-91***	Solidification (bitumen)	Discontinued. 3/4/88
VIKEM	WM-13	Solidification/oil (cement)	Discontinued. 3/9/87
Stock	WM-92***	Solidification (cement)	Discontinued. 6/24/88
Nuclear Packaging	WM-71	Solid/Encap (cement/gypsum)	Withdrawn. 11/21/85
LN Technologies	WM-57	HIC (polyethylene)	Withdrawn. 5/13/85
Chem-Nuclear	WM-47	HIC (fiberglass/poly)	Withdrawn. 5/2/86
Chem-Nuclear	WM-19***	Solidification (cement)	Withdrawn. 10/87
Chem-Nuclear	WM-96***	Solidification (cement)	Withdrawn. 5/27/88
Hittman	WM-79***	Solidification (SG-95)	Withdrawn. 6/10/88
			Submitted
Chem-Nuclear	WM-101	Solidification (cement #1)	Under review. 6/1/88
Chem-Nuclear	WM-97	Solidification (cement #2)	Under review. 6/3/88
Chem-Nuclear	WM-98	Solidification (cement #3)	Under review. 6/10/88
LN Technologies	WM-20	Solidification (cement)	Under review. 6/6/84
LN Technologies	WM-99	Solidification (cement/decon)	Under review. 7/22/88
Hittman	WM-46	Solidification (cement)	Under review. 4/10/84
ATI	WM-100	Solidification (bitumen)	Under review. 8/1/88
Chem-Nuclear	WM-18	HIC (polyethylene)	Under review. 12/29/83
Hittman	WM-80	HIC (polyethylene)	Under review. 6/28/84
TFC	WM-76	HIC (polyethylene)	Under review. 6/26/84
Nuclear Packaging	WM-87	HIC (316-stainless)	Under review. 8/84
LN Technologies	WM-93	HIC (stainless/poly)	Under review. 9/11/87
Bondico	WM-94	HIC (fiberglass/poly)	Under review. 2/26/88
Babcock & Wilcox	WM-95	HIC (coated carbon steel)	Under review. 4/21/88

* Approved for single waste stream for one year.
** Approved pending satisfactory completion of thermal cycling tests.
*** Actions completed in Calendar Year 1988.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555

September 16, 1988

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: SUITABILITY OF HIGH DENSITY POLYETHYLENE HIGH INTEGRITY CONTAINERS

During the fourth meeting of the Advisory Committee on Nuclear Waste, September 13-14, 1988, we met with the Low-Level Waste Management staff and reviewed the status of the staff's investigation into the suitability of high integrity containers (HICs) constructed from high density polyethylene (HDPE) for Class B or Class C low-level waste. This topic was also discussed during other ACNW meetings. The most recent reviews were held during the first meeting of the ACNW on June 28, 1988 and during the field trip to South Carolina, which was held in conjunction with the ACNW's third meeting on August 3-5, 1988. We also had the benefit of the documents referenced.

The Committee heard a well-structured presentation on the technical issues concerning the suitability of HDPE HICs for the disposal of low-level radioactive waste. The focal points of the presentation were the mechanical properties of the present designs and the ability of these designs to meet the NRC requirements for a satisfactory waste container. The staff had obtained expert technical opinion on the pertinent topics and had made effective use of dialogue among knowledgeable parties.

On the basis of the information presented to the Committee, it appears that the present designs of HDPE HICs will have difficulty in meeting the NRC criteria that define their mechanical properties for use as containers for Class B or Class C waste. We are mindful of HDPE's low corrosion rates which, when coupled with other materials that provide the necessary mechanical properties, could result in a container that should be able to satisfy the pertinent NRC criteria. Thus, we have not heard information that would eliminate HDPE from consideration as part of an HIC.

We recommend that the staff bring to closure its study of the HDPE HICs whose designs have been submitted to it for approval. We believe that

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staff decisions would then allow the industry to better plan its response and further action, if any.

Sincerely,

Dade W. Moeller

Dade W. Moeller
Chairman

References:

1. Engineering Design and Testing Corporation Report, submitted to NUS July 21, 1986, "An Assessment of Polyethylene as a Material for Use in High Integrity Containers"
2. U.S. Nuclear Regulatory Commission draft report dated April 6, 1987, prepared by J. Pires, Brookhaven National Laboratory, "Review of the High Integrity Cask Structural Evaluation Program"
3. Letter dated February 2, 1988 from David G. Ebenhack, Chem-Nuclear Systems, Inc., to M. Tokar, NMSS, NRC, attaching Chem-Nuclear Systems, Inc. report dated January 29, 1988, "Evaluation of Stress Loadings of CNSI HDPE HICS"
4. Memorandum dated June 15, 1988 from M. Tokar, NMSS, NRC, to S. J. Parry, ACRS, transmitting U.S. Nuclear Regulatory Commission, Division of Low-Level Waste Management and Decommissioning Report dated June 10, 1988, prepared by S. A. Silling, Brown University, "Review of the Structural Designs of Polyethylene High Integrity Containers"

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

ACTION

DR
DA
JC

November 16, 1990

I think its practical to waive review - please advise by 12/7 ELJ

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

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

SUBJECT: WAIVER OF CRGR REVIEW OF PROPOSED GENERIC LETTER ON THE REMOVAL OF COMPONENT LISTS FROM TECHNICAL SPECIFICATIONS

For recent operating licenses, the NRC has issued Technical Specifications (TS) without the tables that list components to which various specifications apply. These TS follow the principles established by Generic Letter (GL) 84-13 that provided guidance on the removal of the list of snubbers from TS. The principles of GL 84-13 include (1) stating TS requirements in terms that specifically include those components contained on the lists removed from the TS, (2) confirming that these component lists are included in plant procedures, and (3) controlling changes to the component lists by means of the TS administrative control requirements for changes to plant procedures.

Licensees for some plants have included the component lists in the Updated Safety Analysis Report (USAR). Any change to correct or update component lists in the USAR is subject to the provisions of 10 CFR 50.59. This alternative is another means by which licensees may control changes to component lists without processing a license amendment, as is required when the lists are included in the TS.

Enclosure 1 is a proposed generic letter to provide guidance on a license amendment request to remove component lists from plant TS. This TS change is being proposed as a line-item TS improvement. Enclosure 2 is a draft memorandum that provides instructions to project managers on processing license amendments to implement the TS changes. Enclosure 3 is a model safety evaluation report (SER) for these license amendments. Because the proposed action involves a change to the guidance provided by the Standard Technical Specifications, it is subject to CRGR approval. However, we recommend that CRGR waive review of this proposal for the following reasons:

1. The changes described in the proposed generic letter do not alter TS requirements that apply to the components that are individually listed in TS tables.

2. This action is consistent with current practice and does not represent a new staff position.

3. Any proposal by a licensee to implement this TS change is voluntary.

Contact: T. Dunning, OTSB/DOEA
X21189

~~9011300303~~ BPP

A response to our recommendation for waiving CRGR review is requested at your earliest convenience. If you find that CRGR review of this action is necessary, we will prepare a package for CRGR review. This action is sponsored by Charles E. Rossi, Director, Division of Operational Events Assessment.

Frank J. Miraglia

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

Enclosure:
As stated

UNITED STATES
NUCLEAR REGULATORY COMMISS ON
WASHINGTON, D. C. 20555

TO ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PE ITS FOR NUCLEAR POWER REACTORS

SUBJECT: REMOVAL OF COMPONENT LISTS FROM TECHNICAL SPE IFICATIONS (Generic Letter 90-)

This generic letter provides guidance for preparing a r quest for a license amendment to remove component lists from Technical Spec fications (TS). This guidance provides an acceptable alternative to id tifying every component by its plant identification number as curren y exists in tables of TS components. The removal of component lists is a ceptable because it does not alter existing TS requirements or those comp nents to which they apply. The nuclear industry and the NRC identified this line-item TS improvement during investigations of TS problems. Previous guidance was provided by Generic Letter 84-13 on removing the list of snubbers from TS.

This guidance includes the incorporation of lists into plant procedures that are subject to the change control provisions for plant procedures in the Administrative Controls Section of the TS. The removal of component lists from TS permits administrative control of changes to these lists without processing a license amendment, as is required to update TS component lists. Any change to component lists contained in plant procedures is subject to the requirements specified in the Administrative Controls Section of the TS on changes to plant procedures. Therefore, the change control provisions of the TS provide an adequate means to control changes to these component lists, when they exist in or have been incorporated into plant procedures, without including them in TS.

Licensees and applicants are encouraged to propose TS changes that are consistent with the guidance provided in Enclosure 1. The NRC project manager for the facility will review conforming amendment requests. Proposed amendments that deviate from this guidance will lengthen review time. Please contact the project manager or the contact identified below if you have questions on this matter.

Thi letter does not require any licensee to implement changes to their plant procedures or propose changes to their plant TS. Therefore, any action taken in response to the guidance provided in this generic letter is voluntary and is not a backfit under 10 CFR 50.109.

However, the staff is treating this guidance as a request for information. This request relates to TS changes requested by licensees, which is already covered by Office of Management and Budget Clearance Number 3150-0011, which

Contact: Tom Dunning, NRR/OTSB
(301) 492-1189

expires January 31, 1991. The estimated burden hours are 50 person-hours per owner response, including assessment of the staff recommendation and preparing the license amendment application. The estimated burden hours pertain only to the identified response-related matters and do not include the time for actual implementation of the requested action. This generic letter does not alter the burden-hours associated with preparation of similar TS changes and license amendment application. Send comments regarding this burden estimate or any other aspect of the collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch (MNBB-7714), Division of Information Support Services, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555; and to the Paperwork Reduction Project (3150-0011), Office of Information and Regulatory Affairs, NEOB-3019, Office of Management and Budget, Washington, DC 20503.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Removal of Component Lists from
   Technical Specifications
2. List of Recently Issued Generic Letters

REMOVAL OF COMPONENT LISTS FROM TECHNICAL SPECIFICATIONS (TS)

Background:

Generic Letter (GL) 84-13 provided guidance on removing the list of snubbers from Technical Specifications (TS). After GL 84-13 was issued, many licensees submitted proposals on a plant-specific basis to remove other component lists from TS. The nuclear industry has also recommended the removal of component lists from TS as a TS improvement. This guidance for a license amendment request to remove component lists from TS is based on the experience of both the NRC and the industry.

The NRC staff noted that many license amendments had been required to add, delete, or modify the list of snubbers. The staff concluded that the list of snubbers was not necessary, provided the TS were modified to specify those snubbers that are required to be operable. Also, the staff noted that any changes in the quantities, types, or locations of snubbers would constitute a change to the facility and thus would be subject to the provisions of 10 CFR 50.59. The snubber TS was modified to state that the only snubbers excluded from the TS requirements were those installed on nonsafety-related systems, and then only if their failure or the failure of the system on which they were installed would have no adverse effect on any safety-related system. The table with the list of snubbers and the associated references were removed from the Limiting Condition for Operation (LCO) and the associated surveillance requirements.

Therefore, specifications may be stated in general terms that describe the types of components to which the requirements apply. This provides an acceptable alternative to identifying components by their plant identification number as currently exists in tables of TS components. The removal of component lists is acceptable because it does not alter existing TS requirements or those components to which they apply.

Guidance on the Removal of Component Lists From TS:

The approach taken in GL 84-13 to remove a list of components from TS may also be used to remove other component lists from TS. To implement this approach, the TS should be revised to incorporate an explicit description of those components for which the TS requirements apply. A list of those components must be included in a plant procedure that is subject to the change control provisions for plant procedures in the Administrative Controls Section of the TS. This can be accomplished by incorporating the list, that identifies all the components for which the TS requirements apply, in such procedure or by confirming that an existing procedure includes this list of components. When the component list is included in a plant procedure, the identification of the individual components to which the TS requirements apply will be a simple task.

Although some components may be listed in the updated safety analysis report (USAR), the USAR should not be the sole means to identify these components. Licensees are only required to update the USAR annually, and they are only required to reflect changes made 6 months before the date of filing. Thus, the USAR may be out of date by as much as 18 months. However, to highlight the change controls of 10 CFR 50.59 or to clarify other issues related to these components, licensees may wish to include these component lists in the next update of the USAR. The Bases Section of the TS may reference the plant procedures where these lists are located; however, component lists should not be included in the Bases Section because the Bases Section lacks an appropriate regulatory process for change control.

The staff provides the following guidance for changing individual TS sections. This guidance addresses considerations unique to specific types of component lists.

1. Containment Isolation Valves

The specification for containment isolation valves applies to those valves that are listed in the table referenced in the TS. The alternative to listing these valves in a TS table is the revision of the LCO to state "Each containment isolation valve shall be OPERABLE." Similarly, the surveillance requirements for (1) post-maintenance testing, (2) demonstrating automatic closure on isolation signals, and (3) confirming the isolation time of power-operated or automatic valves, should be revised to remove the reference to the TS table and revised to state "Each containment isolation valve shall . . ." or ". . . each power-operated or automatic containment isolation valve shall . . ."

The list of containment isolation valves in the TS may not include all valves that are classified as containment isolation valves by the plant licensing basis. Generally, the USAR identifies those valves that are classified as containment isolation valves. With this TS change, the LCO, remedial action and surveillance requirements will apply for all valves that are classified as containment isolation valves by the plant licensing basis.

The list of containment isolation valves typically includes notes that modify the TS requirements for these valves. Such notes must be incorporated into the associated LCO so that these notes will remain in effect when the table containing these notes is removed from the TS. One of these notes involves valves that are exempt from the requirements of Specification 3.0.4. Specification 3.0.4 precludes entry into an operational mode or condition when an LCO would not be met without reliance on the provisions of the action requirements. The action requirements for containment isolation valves permit continued operation with an inoperable valve when the associated penetration is isolated. Therefore, an exception to the limitation of Specification 3.0.4 on changes in operational modes or conditions is acceptable for this TS, and a footnote may be added to the LCO to state "The provisions of Specification 3.0.4 do not apply." The exception, provided by this footnote, will now be applicable to all containment isolation valves. The increase in the scope of this exception is acceptable because it is consistent with the guidance provided in Generic Letter 87-09. However, this footnote is not necessary if Specification 3.0.4 has been revised as allowed by Generic Letter 87-09.

The list of containment isolation valves may also include a note that clarifies an operational consideration for specific valves that may be opened on an intermittent basis under administrative control. This clarification applies to local manually-operated valves that are locked or sealed closed consistent with the design requirements of General Design Criteria 55, 56, and 57 of Appendix A to 10 CFR Part 50. The design of these valves includes positive control features to ensure that they are maintained closed. Therefore, opening locked or sealed closed valves is contrary to the operability requirements for these valves that are currently listed in the TS table of containment isolation valves. With the removal of this list of valves, the TS operability requirements will apply to all local manual-operated locked or sealed closed containment isolation valves. The staff concludes that an acceptable alternative to identifying specific valves that may be opened under administrative control would be a footnote to the LCO to state "Local manual-operated locked or sealed closed valves may be opened on an intermittent basis under administrative control." With this change, the definition of Containment Integrity and the surveillance requirements for demonstrating containment integrity in Specification 4.6.1.1 should be revised to remove the reference to the table of containment isolation valves. These sections of the TS will then just reference the containment isolation valve specification that identifies the exception that is addressed by the new footnote on opening valves on an intermittent basis under administrative control.

The note on opening valves under administrative control also may have been used in some plant TS for remote-manual valves in closed systems inside containment. A remote-manual valve is an acceptable alternative to a locked or sealed closed valve for a closed system inside containment as noted in General Design Criterion 57 in Appendix A to 10 CFR Part 50. Therefore, this note need not remain in the TS to allow operators to open any remote-manual containment isolation valve because such action is not contrary to the operability requirements for these valves.

Another clarifying note used in the list of containment isolation valves identifies those valves that are not subject to Type C leak testing requirements of Appendix J to 10 CFR Part 50. In this case, this notation does not alter the requirements of Appendix J but rather only clarifies where the NRC has granted exemptions to Type C leak testing or where Addendix J does not require this testing. Therefore, the TS need not include this clarification, but it may be included with a list of these valves in the USAR if desired to clarify the applicability of Appendix J requirements. However, placing the list of containment isolation valves currently in TS in the USAR would not restrict the applicability of the TS requirements to only the valves on that list. As previously noted, the TS requirements would apply to all valves that have been defined as containment isolation valves in the plant licensing basis.

Finally, some TS have included valve closure times in the list of containment isolation valves. The inservice testing (IST) requirements referenced by Specification 4.0.5 include the verification of valve stroke times for a broader class of valves than those containment isolation valves that have been listed in the TS. The removal of valve closure times that are included in some plant TS would not alter the IST requirements to verify that valve stroke times are within their limits; and therefore, removal of these closure times is acceptable.

Because plant-specific considerations may have required that these tables include other notes modifying the TS requirements for specific valves, any such exceptions should be stated in terms that identify the valves by function rather than by component number, if practical. This guidance also applies to any other component list removed from TS that includes notes that alter the

TS requirements. If notes in these tables are only included for information or clarification and do not alter any TS requirement, the removal of these notes with the list of components would not affect the applicability of the TS requirements.

2. Reactor Coolant System Pressure Isolation Valves

Guidance on removing from the TS the list of reactor coolant system pressure isolation valves is pending the NRC staff's resolution of generic concerns with existing lists for these valves. In the interim, licensees should not submit proposals to remove this list from the TS.

3. Secondary Containment Bypass Leakage Paths

The TS on containment leakage include a list of secondary containment bypass leakage paths. The list identifies these leakage paths by penetration number for dual containment plants. The combined leakage rate for all penetrations identified as secondary containment bypass leakage paths is specified.

As part of the plant licensing basis, the USAR defines the penetrations that are secondary containment bypass leakage paths. This definition of "secondary containment bypass leakage paths" is adequate such that the TS requirements do not require further clarification upon the removal of this list from the TS. Therefore, the TS requirements may be stated in terms of secondary containment bypass leakage paths without further clarification. For example, the limitation of TS 3.6.1.2.c on containment leakage rates should be revised to state the following:

> A combined leakage rate of less than or equal to [0.10] La for all penetrations that are secondary containment bypass leakage paths when pressurized to Pa.

4. Containment Penetration Conductor Overcurrent Protective Devices

The list of containment penetration conductor overcurrent protective devices includes those primary and backup fuses and breakers that preclude faults of a magnitude and duration that could compromise the integrity of electrical penetrations. Because the number of overcurrent protective devices associated with electrical circuits penetrating containment may exceed the basic requirements for primary and backup protection, the description of these components should be stated to clarify those components to which the TS requirements apply. Also, these requirements exclude circuits for which credible fault currents would not exceed the electrical penetration design rating. For example, these requirements exclude thermocouple and other low-power-level signal circuits. An alternative to listing these components in a TS table is the following LCO statement:

> Primary and backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes

those circuits for which credible fault currents would not exceed the electrical penetration design rating.

In addition, the surveillance requirements should state "The above noted primary and backup containment penetration conductor overcurrent protective devices . . ." rather than referring to those components listed in Table 8.3-1.

5. Motor-Operated Valves Thermal Overload Protection

The TS contain a list of valves that have thermal overload protection and bypass devices integral with the motor starter. The table in the TS lists the valves by number, the bypass device, and the system affected. With the removal of this list of valves from the TS, the LCO should state "The thermal overload protection and bypassed devices, integral with the motor starter, of each valve used in safety systems shall be OPERABLE." This statement for the LCO adequately defines the scope of the valves that include these features to which the TS requirements apply.

6. Other Component Lists

Component lists other than those previously described herein may be candidates for removal from TS on a plant-specific basis. A proposal to remove other component lists from TS should be based on this guidance and any specific considerations applicable to each list.

Summary:

In summary, a request to remove component lists from TS should address the following issues:

1. Each TS should include an appropriate description of the scope of the components to which the TS requirements apply. Components that are defined by regulatory requirements or guidance need not be clarified further. However, the Bases section of the TS should reference the applicable requirements or guidance.

2. If the removal of a component list results in the loss of notes that modify the TS requirements, the specification should be changed to incorporate the specific modification or exception to the requirements. The exception should be stated in terms that identify the valves by function rather than by component number, if practical.

3. Licensees should confirm that the lists of components removed from the TS are located in appropriately controlled plant procedures. The list of components may be included in the next update of the USAR. The Bases of the individual specifications also may reference controlled plant procedures or other documents that identify each component list.

This guidance should not be used to remove tables from TS that address information or requirements other than the lists of components to which a specification applies.

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

Enclosure 2

MEMORANDUM FOR: All NRR Project Managers

FROM: James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

SUBJECT: GENERIC LETTER 90-

Enclosure 1 is Generic Letter 90- which provides guidance to licensees for a license amendment request to remove component lists from Technical Specifications (TS). Any proposal for this line-item TS improvement is voluntary.

Project managers should perform the review and process proposed license amendments conforming to the guidance of the generic letter. Generally, the project managers need not consult or obtain review assistance from a technical review branch unless the proposed amendment deviates from the generic letter guidance.

Enclosure 2 is a model safety evaluation report (SER) that was prepared by the Technical Specifications Branch. This model SER should assist you in your preparation of a license amendment to implement this line-item TS improvement. The lead project manager for this task is _____ _____. _____ will assist you in the preparation of a no-significant hazards consideration pre-notice for a proposed amendment conforming to the generic letter and should be included on distribution for the amendment package.

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
Generic Letter 90-
Model SER

cc w/enclosures:
J. Sniezek
H. Thompson
Division Directors, NRR
Associate Directors, NRR
Project Directors, NRR
Regional Administrators
J. Conran, CRGR
C. Berlinger, DOEA
S. Treby, OGC

CONTACT:
T. L.nning, OTSB, NRR
492-1189

Enclosure 3

MODEL SAFETY EVALUATION REPORT

Underscored blank spaces are to be filled in with the applicable information. The information identified in brackets should be used as applicable on a plant-specific basis.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO.__ TO FACILITY OPERATING LICENSE NFP-__
AND AMENDMENT NO.__ TO FACILITY OPERATING LICENSE NFP-__
[UTILITY NAME]
DOCKET NOS. 50-__ AND 50-__
[PLANT NAME], UNITS 1 AND 2

INTRODUCTION

By letter of _____ __, 1990, [utility name] (the licensee) proposed changes to the Technical Specifications (TS) for [plant name]. The proposed changes remove tables providing lists of components referenced in individual specifications. In addition, the TS requirements have been modified such that all references to these tables have been removed. Finally, the TS requirements have been modified to state the requirements in general terms that include the components listed in the tables removed from the TS. Guidance on the proposed TS changes was provided by Generic Letter 90- , of _____ __, 1990.

EVALUATION

The licensee has proposed the removal of Table 3.6-1, "Secondary Containment Bypass Leakage Paths," that is referenced in TS 3.6.1.2. With the removal of this table, the licensee has proposed to modify the limiting condition for operation (LCO) on containment leakage rates to state the limit specified by TS 3.6.1.2.c as the following:

> A combined leakage rate of less than or equal to [0.10] La for all penetrations that are secondary containment bypass leakage paths when pressurized to Pa.

The licensee has proposed the removal of Table 3.6-[2], "Containment Isolation Valves," that is referenced in TS 3/4.6.4. With the removal of this table, the licensee has proposed to include the following statement of the LCO under TS 3.6.4:

> Each containment isolation valve shall be OPERABLE.

In addition, the licensee has revised the definition of Containment Integrity, TS 4.6.1.1 and 4.6.4.1 through 4.6.4.3 to remove the reference to Table 6.3-[2]. The definition of Containment Integrity and TS 4.6.1.1 refer to TS 6.6.4 for an exception that is now covered by a footnote to the LCO rather than by the table removed from the TS. The surveillance requirements of TS 4.6.4.1 through 4.6.4.3 have been revised to state "Each containment isolation shall. . ." or ". . . each power-operated or automatic containment isolation valve shall . . ." rather than stating the requirements in relation to the valves specified in Table 3.6-[2]. [Because Table 3.6-[2] notes that the provisions of Specification 3.0.4 are not applicable to specific valves, the following footnote has been added to the LCO for TS 3.6.4:

The provisions of Specification 3.0.4 do not apply.

This is a change in the scope for this exception, from specific valves to all containment isolation valves and is acceptable because it is consistent with the guidance provided in Generic Letter 87-09 as noted in Generic Letter 90- .]

The table of containment isolation valves identified specific local manual-operated locked and sealed closed valves with a footnote stating that these valves may be opened on an intermittent basis under administrative control. These valves are locked or sealed closed consistent with the regulatory requirements for local manual-operated valves that are used as containment isolation valves. Because opening these valves would be contrary to the operability requirements of these valves, the following footnote to the LCO has been proposed:

> Local manually-operated locked or sealed closed valves may be opened on an intermittent basis under administrative control.

This change is consistent with the guidance in Generic Letter 90- and is, therefore, acceptable.

The licensee has proposed the removal of Table 3.6-1, "Containment Penetration Conductor Overcurrent Protective Devices" that is referenced in TS 3/4.8.4.2. With the removal of this table, the licensee has proposed to include the following statement for the LCO under TS 4.8.3.2:

> Primary and backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes those for which credible fault currents would not exceed the electrical penetration design rating.

In addition, the licensee has proposed to revise TS 4.8.3.2 to remove the reference to Table 8.3-1. The surveillance requirement has been revised to state the following:

> The above noted primary and backup containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

The licensee has proposed the removal of Table 3.8-2, "Motor-Operated Valves Thermal Overload Protection," that provides a list of valves with bypass devices that is referenced in TS 3.8.4.3. With the removal of this table, the licensee has proposed to include the following statement of the LCO under TS 3.8.3.3:

> The thermal overload protection and bypass devices, integral with the motor starter, of each valve used in safety systems shall be OPERABLE.

The licensee has proposed changes to the above TS that are consistent with the guidance provided in Generic Letter 90- . [In addition, the licensee has proposed changes to TS 3.6.4 such that exceptions to the requirements of the LCO that were included in the table that has been removed are now addressed by a footnote to the action requirements.] Finally, the licensee has confirmed that the list of components included in the tables removed from the TS are located in controlled plant procedures. [This list of components will also be included in the next revision of the Updated Safety Analysis Report.] (NOTE to PMs: The inclusion of this list in the next USAR update is not a requirement, but the SER should reflect any commitment by the licensee to do so.)

On the basis of its review of this matter, the staff finds that the proposed changes to the TS for (plant name) Unit(s) ___ are an administrative change that does not alter the requirements set forth in the existing TS. However, this change will allow licensees to make corrections and updates to the list of components for which those TS requirements apply, under the provisions that control changes to plant procedures as specified in the Administrative Controls Section of the TS. Therefore, the staff finds that the proposed TS changes are acceptable.

ENVIRONMENTAL CONSIDERATION

This (These) amendment(s) involve changes in recordkeeping, reporting, or administrative procedures or requirements. The amendment(s) remove lists of components which are subject to the TS requirements for limiting conditions for operation (LCOs) and surveillances, and includes them in controlled plant procedures. Accordingly, the amendment(s) meet(s) the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Existing TS requirements with regard to LCOs and surveillances are not changed by the removal of the component lists. Since the component lists are located in controlled plant procedures, any changes or corrections to these lists must be made in a controlled manner as specified in the Administrative Controls Section of the Technical Specifications. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this (these) amendment(s).

CONCLUSION

The Commission made proposed determinations that the amendment(s) involve no significant-hazards consideration, which were published in the Federal Register (5_ FR _____) on ______ __, 199_. The Commission consulted with the State of ______. No public comments were received, and the State of ______ did not have any comments.

On the basis of the considerations discussed herein, the staff concludes that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Thomas G. Dunning, OTSB/DOEA
_______________, PD__/DRP__

Dated: _______ __, 199_

(Note to PM's: A copy of this document may be obtained from P. Coates, X-21161, by requesting 5520 document: "LIST SER." It can be transmitted electronically to your secretary or licensing assistant.)