Agenda
NRC/SCANA/Dominion Counterparts Meeting
May 9-10, 2000

Location: Innsbrook Technical Center
5000 Dominion Blvd
Glen Allen, VA 23060

Attendees

NRC:	Herbert Berkow -	NRC Project Directorate II - Director
	Richard Emch, Jr. -	NRC Project Directorate II - Section Chief
	Gordon Edison -	NRC Project Manager - Surry
	Steve Monarque -	NRC Project Manager - North Anna
	Karen Cotton-	NRC Project Manager - V. C. Summer
Virginia Power:	Jim McCarthy -	Manager - Nuclear Licensing and Operations Support
	Jimmy Hayes -	Manager - NAPS Safety & Licensing
	Toby Sowers -	Manager - SPS Safety & Licensing
SCANA:	Mel Browne -	Manager - Licensing

Time	Topic	Responsible
May 9th **Private Dining Room**		
1000 - 1015	Introductions / Opening Comments	Berkow/Cotton/ Browne/McCarthy
1015 - 1125	Licensing Process	
	NRC Perspective	Monarque
	Role of the PM	Berkow
	Licensee Perspective & Process	Rose/Sommers
1125 - 1200	ADAMS Project / Electronic Correspondence Demonstration / Feedback	Cotton/ Rosbaugh/McClure
1200 - 1245	Lunch	
1245 - 1345 **Private Dining Room or** **Conf. Room IN2SE-D**	Breakout Session #1 Attributes of a Good Licensing Submittal Attributes of a Good SER	 NRC SCANA/VP
1345 - 1415 **Private Dining Room**	Joint Wrap-Up Discussion Attributes of a Good Licensing Submittal	NRC SCANA/VP
1415 - 1430	Break	

1430 - 1530	Discussions on Streamlining the Licensing Process	
	Tech Spec Line Item Improvement (CLIIP)	NRC
	Cost Reporting / Understanding	SCANA
	Cost versus Risk Significance	SCANA
	Cost Control & Real Time Monitoring	SCANA
	Resident Review of Routine Items	SCANA
	Joint Submittal Reviews / Processing	VP
	Priority versus Backlog Considerations	VP
	Refueling Outage Submittal Expediting	SCANA
	Tech Spec Bases Changes / Commitment Changes	VP
	Administrative Corrections of Tech Specs	SCANA
1530 - 1630	PRA Capabilities & Applications	Estes/Bucheit

May 10th

0800 - 0900 **Private Dining Room or Conf. Room IN2NE-D**	Breakout Session #2 Assessment of License Amendments Review of Two Examples	NRC SCANA/VP
0900 - 0930 **Private Dining Room**	Joint Wrap-Up Discussion Assessment of License Amendments	NRC SCANA/VP
0930 - 0945	Break	
0945 - 1100	Regulatory Issues/Initiatives	
	Design Basis Definition	Cotton
	UFSAR Updating	Monarque
	10 CFR 50.72/73 Reporting	Cotton
	10 CFR 50.59	Emch
	License Renewal	Berkow
	Weather-Related NOEDs	Berkow
1100 - 1130	Attributes of a Good Relief Request	Edison
1130 - 1200	Criteria for Environmental Assessments	Edison
1200 - 1215	Concluding Comments	Berkow/Browne/ McCarthy

LICENSING WORKSHOP

Office Letter 803

STEPHEN R. MONARQUE

PROJECT MANAGER

DIVISION OF LICENSING PROJECT MANAGEMENT

OFFICE OF NUCLEAR REACTOR REGULATION

May 9, 2000

OFFICE LETTER 803

- WORK PLAN
- PUBLIC NOTIFICATION
- SAFETY EVALUATION
- AMENDMENT ISSUANCE

WORK PLAN

- REVIEW APPLICATION FOR COMPLETENESS AND ACCEPTABILITY
  - ◆ OATH & AFFIRMATION
  - ◆ CLEAR DESCRIPTION OF AMENDMENT
  - ◆ REQUESTED REVIEW SCHEDULE
  - ◆ TECHNICAL SPECIFICATION PAGES

WORK PLAN (continued)

- REVIEW APPLICATION FOR COMPLETENESS AND ACCEPTABILITY (continued)
  - SAFETY ANALYSIS AND JUSTIFICATION FOR PROPOSED CHANGE
  - NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)
  - SEARCH FOR PRECEDENTS

WORK PLAN (continued)

- PRIORITY OF LICENSE AMENDMENTS
  - SAFETY CONCERNS
  - PLANT SHUTDOWN, OR RESTART
  - RISK INFORMED LICENSING ACTION
  - MAINTAIN SAFE PLANT OPERATIONS
  - COST BENEFICIAL LICENSING ACTIONS

PUBLIC NOTIFICATION

- 10 CFR 50.91

- 30 DAY PUBLIC NOTIFICATION, 10 CFR 50.91(a)(2)
  - NSHC
  - SEEKS PUBLIC COMMENT FOR 30 DAY TIME FRAME
  - FEDERAL NOTICE PUBLISHES PROPOSED LICENSE AMENDMENT AND NSHC DETERMINATION

PUBLIC NOTIFICATION (continued)

- EMERGENCY AND EXIGENT PUBLIC NOTIFICATIONS
  - REQUIRES JUSTIFICATION AND NSHC
  - EXIGENT 10 CFR 50.91(a)(6)
    - SEVEN TO THIRTY DAY TIME FRAME
    - TWO WEEK COMMENT PERIOD

PUBLIC NOTIFICATION (continued)

- EMERGENCY AND EXIGENT PUBLIC NOTIFICATIONS (CONTINUED)
  - EMERGENCY 10 CFR 50.91(a)(5)
    - NOTICE OF LICENSE AMENDMENT PROVIDES OPPORTUNITY FOR COMMENTS

SAFETY EVALUATION

- TECHNICAL SAFETY AND LEGAL BASIS OF AN AMENDMENT REQUEST
  - STATE CONSULTATION
  - FINAL NSHC DETERMINATION INCLUDED

- REQUESTS FOR ADDITIONAL INFORMATION

AMENDMENT ISSUANCE

- OBTAIN STAFF CONCURRENCE

- SUBMIT NOTICE OF LICENSE AMENDMENT TO FEDERAL REGISTER

- CONTACT STATE FOR ANY COMMENT TO NSHC

AMENDMENT ISSUANCE (continued)

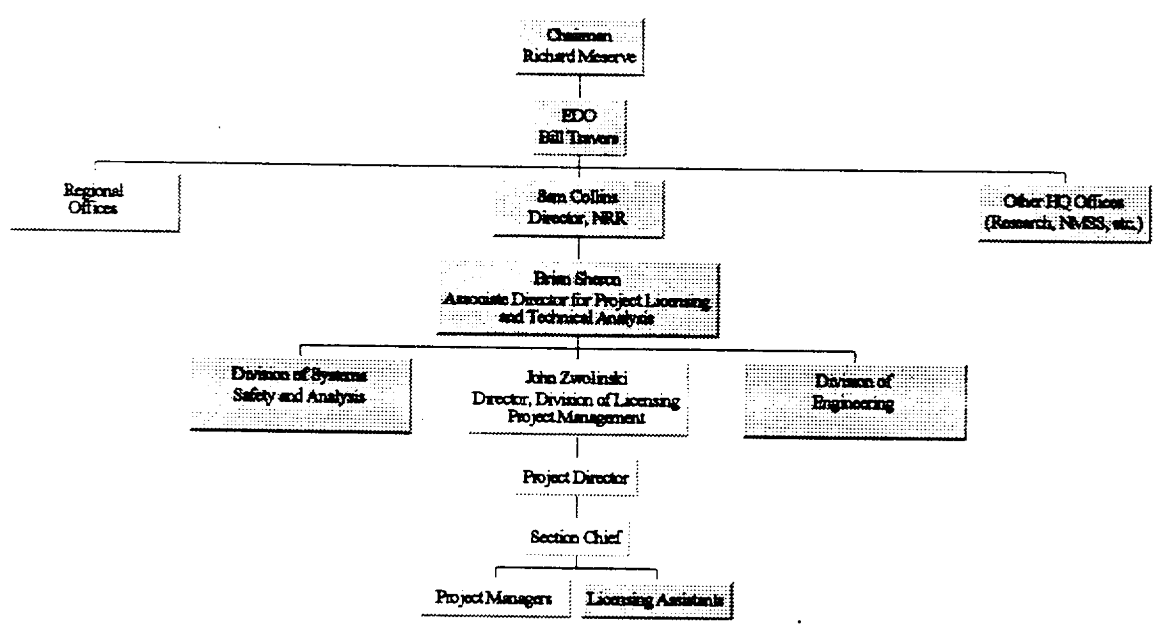
- DETERMINE IF PUBLIC COMMENTS OR PETITIONS HAVE BEEN RECEIVED

- SEND LETTER TO LICENSE WITH FOLLOWING ENCLOSURES
  - REVISED TECHNICAL SPECIFICATION PAGES
  - INPUT TO FEDERAL REGISTER
  - SAFETY EVALUATION WITH EA, IF APPROPRIATE

DLPM PROJECT MANAGER RESPONSIBILITIES

Herb Berkow
Director - Project Directorate II
Division of Licensing Project Management

DLPM ORGANIZATION



BACKGROUND

- Generally, one Project Manager per site
- PM assignments are for a maximum of 5 years
- Educational background is typically engineering
- Experience is varied (nuclear industry, regional inspectors, other NRC offices)

EXPECTATIONS

- Most knowledgeable member of the staff regarding the licensing agenda for assigned facility
- Knowledgeable of plant design and operation
- Thorough understanding of NRC rules, processes and licensing requirements
- Focal Point for NRC/Licensee Correspondence
- Prioritize, Schedule, Review, Manage & Prepare all actions associated with the licensing process
- Maintain NRC information management systems

PERFORMANCE MEASURES

- Timeliness
- Effectiveness
- Efficiency
- Quality
- Quantity

STRATEGIC OUTCOME GOALS

- Maintain Safety
- Reduce Unneccessary Regulatory Burden
- Increase Public Confidence
- Increase Internal Efficiency & Effectiveness

DLPM IMPLEMENTING PLAN

- Licensing Authority
- Interfaces
- Regulatory Improvements
- Total of 75 Specific Tasks

LICENSING AUTHORITY

- Licensing Actions
- Mandated Controls
- Other Licensing Tasks

LICENSING ACTIONS

- License Amendments (TS & USQ)
- Exemptions
- Relief Requests
- License Transfers
- NOEDs
- Lead Plant Reviews

MANDATED CONTROLS

- TS Bases Changes
- UFSAR Reviews (10 CFR 50.71(e))
- Facility Changes (10 CFR 50.59)
- QA, Security, EP Program Reviews

OTHER LICENSING TASKS

- Pre-Application Reviews
- Task Interface Agreements
- 10 CFR 2.206 Petitions
- Plant-Specific Multi-Plant Actions
- Commitment Management
- Hearing Support
- Backfits
- Proprietary Information Reviews

INTERFACES

- Licensee/Owners' Groups
- NRC Headquarters
- Regional Offices
- Public

LICENSEE/OWNERS' GROUP ACTIVITIES

- Routine Communications with Licensee
- Site Visits/Drop-ins
- Lead PM on Technical Issues (MPAs, GSIs, USIs)

HEADQUARTERS

- Management Info. & Status Reports
- Incident Response
- Miscellaneous Licensee Reports
- Fee Billing Reviews
- Surveys
- General Support to other NRC Offices

REGIONS

- Morning Plant Status Calls
- Management Oversight Panels
- Routine Communications
- TS Interpretations
- Enforcement Support
- Event Followup

PUBLIC

- Controlled Correspondence
- Noticing Amendments, meetings
- Allegations
- FOIA requests
- Plant Information on NRC Web page

REGULATORY IMPROVEMENTS

- Licensing Action Task Force
- Owners' Group Interactions
- NRR Office Letters
- NRR Reinvention Effort
- Rulemaking (Risk Informing Part 50)
- Task Forces (ADAMS, Public, Y2K)
- Licensing Workshops

LICENSE AMENDMENT PROCESS

Presented by

Phil Rose

V. C. Summer Nuclear Station

May 9, 2000

License Amendment Requests

- TSCRs
- Preparation/Review
- Implementation

TSCR Preparation

- Initialization and Evaluation
  - Submitted to NL&OE
    - Either formally or by verbal request
  - Evaluated for need and completeness
  - Open tracking document
    - Status, Due Dates, Actions, Commitments

TSCR Development

- Resources
  - Sponsor Group and Licensing
- Possible Overview to PSRC
  - Conceptual with specific changes identified
- Communication with PM

TSCR Development

- Develop Marked-Up pages, Bases change, Description of Changes, Safety Evaluation, NSHD, Environmental Assessment, List of Commitments
- Compare change to NUREG1431 and recent industry changes

Plant Review

- Sponsor Organization
  - Technical adequacy
  - Implementation timeframe
- Plant Interface Review
  - Electronic and hardcopy (~26)
  - PSRC
  - NSRC
- Internal Signoff "Chop" Process
  - Affected Management up through VPNO

Responsibilities of Review Organizations

- Technical Content
- Editorial
- Identify Affected Procedures / Documents
- Identify Training Requirements

Implementation

- Pre-approval
  - Communication with PM
    - Resolve Comments and RAIs
    - Coordinate Effective and Implementation Dates
  - Direction to Affected groups to prepare document and training changes/materials

Implementation

- Post-approval
  - Review for correctness
  - Alert Plant
    - Minimum of Operations, Training, Affected Organization(s)
  - Alert Resident Inspector
  - Verify Document Changes / Training completed
  - Issue Amendment at same time as Document Revisions

Implementation

- Exceptions
  - Current procedure is more restrictive (conservative) than Amendment
    - Do not have to revise
  - More restrictive (conservative) change needs to be implemented prior to approval
    - Administrative Letter 98-10
    - Administrative controls

REQUEST FOR ENFORCEMENT DISCRETION PROCESS

Presented By

Phil Rose

V. C. Summer Nuclear Station

May 9, 2000

SCE&G
A SCANA COMPANY

REQUEST FOR ENFORCEMENT DISCRETION PROCESS

- Planning
- Implementation

Planning

- Detailed determination of potential violation and effects
- Determination of Oral or Written
- Compile information
  - 12 points (Administrative Letter 95-05, Revision 2)
- Review for technical accuracy
- Editorial

Planning

- Impact on programs and documents
- Have Contingency Plan
- Present to PSRC
  - Concurrence
  - Editorial

Implementation

- DRY RUN
- Make phone call
  - Most knowledgeable persons
  - Senior Plant management
- Answer questions
- FAX to NRR Project Director and Regional Administrator

Prepare Follow-Up Documentation

- Written Request for NOED (24 hours)
- TSCR (48 hours)
- LER

Key Points

- Highest Priority
  - Resources
  - Time
- Have Contingency Plans
- Be sensitive to work schedules
  - NRC Staff
  - Plant Staff
- Practice with plant personnel playing the NRC role

AGENCYWIDE DOCUMENTS ACCESS & MANAGEMENT SYSTEM (ADAMS)

NRC/VEPCO/SCANA WORKSHOP
MAY 9-10, 2000
KAREN COTTON

WHAT IS IT?

- MAINTAIN READ-ONLY RECORDS THAT CAN BE READ FROM MULTIPLE SITES
- FULL TEXT SEARCH CAPABILITY BY NRC AND PUBLIC
- ELECTRONIC DOCUMENTS BECOME OFFICIAL RECORD
- REPLACES NUDOCS

ELECTRONIC INFORMATION EXCHANGE (EIE)

- FUTURE SYSTEM TO PROVIDE ELECTRONIC DOCUMENT EXCHANGE TO AND FROM NRC
- PARTICIPATION IS VOLUNTARY

PARTICIPATION IN EIE

- MUST HAVE ACCESS TO INTERNET VIA INTERNET EXPLORER OR NETSCAPE
- APPLY FOR AND BE GRANTED A "DIGITAL CERTIFICATE".
- 5 MEG (1000 PAGES) LIMIT. LARGER DOCUMENTS WITH PRIOR NOTICE.

PARTICIPATION IN EIE (cont)

- DOCUMENT SUBMITTALS:
  - PDF NORMAL
  - PDF
  - WORD
  - WordPerfect
- MAY BE EXPANDED LATER (ASCII)

EIE PROCESS

- ELECTRONICALLY SIGN DOCUMENT
- PLACE ON EXTERNAL SERVER
- SEND EMAIL TO RECIPIENT
- NO PUBLIC ACCESS TO EIE

EXTERNAL ACCESS

- ACCESS NRC EXTERNAL WEB (NRC.GOV)
- CLICK ON "PUBLIC ELECTRONIC READING ROOM" AT BOTTOM OF PAGE
- FOLLOW INSTRUCTIONS OR CALL LISTED NUMBERS FOR HELP

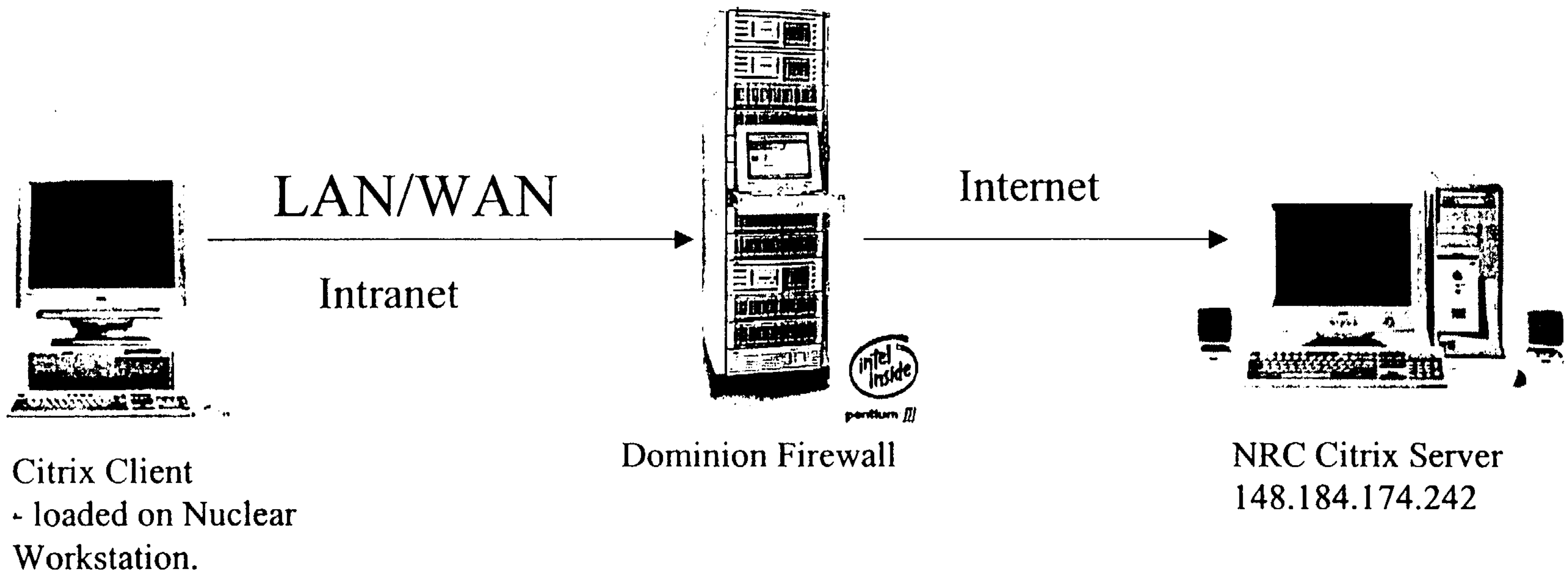
SENSITIVE INFORMATION

- PROPRIETARY, SECURITY, PRIVACY INFORMATION PROTECTED BY ADAMS PROCEDURES AND SOFTWARE
- SAFEGUARDS INFORMATION WILL NOT BE INCLUDED IN ADAMS

NUDOCS

- DOCUMENTS PRIOR TO 11/1/99 WILL CONTINUE TO BE KEPT IN MICROFICHE
- WILL NOT BE CONVERTED TO ADAMS
- CAN SEARCH FOR DOCUMENT BY TITLE IN ADAMS LEGACY LIBRARY

ADAMS Initial Client Connection



A client starts a Citrix session from the client workstation. The NRC Citrix server is contacted through the company firewall through TCP Port 1494 and UDP Port 1604.

RISK

To successfully connect to the NRC Citrix server, the client company must open TCP Port 1494 and UDP Port 1604 on their firewall. Any time a port is open, there is some element of risk involved.

Dominion has minimized this risk by isolating to the Nuclear branch of the Novell Directory Structure tree through the use of Asynchronous Transfer Mode (ATM). We have also configured the ports to accept only Citrix traffic with the NRC Citrix server.

Employee Access

As FOIA information is available through ADAMS, it was determined to open access to all Nuclear Employees by adding a link to the "Industry-Related Web Pages" company intranet page. Since the correct ports are open in the firewall for the entire Nuclear branch, the Citrix client (plugin) will self-install. These users will all log into ADAMS using the guest account.

Limited personnel in Licensing will possess the necessary user identification and passwords to allow data transfer between Dominion and the NRC.

What Dominion Still Needs:

1. Stabilize the ADAMS System. We need to know that actions we take to assure access for employees now will not require changes based on a change at the NRC.

2. Notify us when ADAMS is in full production and all information is real. We do not get the impression now that ADAMS is ready for full-scale use. We do not want to open access for our employees until we are certain that what they see is actual data, not a test in progress.

Consolidated Line Item Improvement Process

Rich Emch
NRC/VEPCO/SCANA
Licensing Workshop

What is CLIIP

- NRC Regulatory Issue Summary 2000-06 issued March 20, 2000
- Describes the CLIIP method for adopting changes to the STS
- Improves efficiency of licensing process by revising TSTF process
- Enhances public visibility of NRC STS revision process by soliciting stakeholder comments

The Process - Generic

- TSTFs are proposed including technical justification and "no significant hazards considerations" information
- Proposed TSTF passes preliminary NRC review
- Public comment solicited by Federal Register Notice
- Public comments resolved
- Approved TSTF including Safety Evaluation & NSHC published on NRC WebSite & in FRN

The Process - Plant Specific

- Licensee submits amendment application referencing approved TSTF
- Consistent with TSTF or providing justifications for necessary deviations
- TSTF NSHC and Safety Evaluation ready to go

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

March 20, 2000

NRC REGULATORY ISSUE SUMMARY 2000-06 CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS FOR ADOPTING STANDARD TECHNICAL SPECIFICATIONS CHANGES FOR POWER REACTORS

ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to inform the addressees of the opportunity to participate as applicants in the consolidated line item improvement process (CLIIP) for Technical Specifications (TS) amendments. The CLIIP facilitates licensees' adopting of NRC-accepted changes to the Standard Technical Specifications (STS) for their specific plant TS. This process is intended to streamline the license amendment review process involving NRC-accepted STS changes in order to increase NRC efficiency and reduce unnecessary regulatory burden. The NRC role in maintaining plant safety is achieved by the technical review of proposed changes to the STS as well as plant-specific applications to adopt NRC-accepted changes to the STS. In addition, the CLIIP is intended to increase public confidence by making NRC's work process more visible to its stakeholders.

The CLIIP improves the efficiency of the NRC licensing processes by reviewing and documenting STS change requests in a manner that supports subsequent license amendment applications. By soliciting comments from NRC stakeholders, the CLIIP enhances the visibility of the staff's review and revision processes for the STS as well as subsequent license amendment applications. Following the staff's resolution of public comments on a proposed change to the STS, the licensees may submit a license amendment application to adopt the NRC-accepted change by citing the relevant information which would have been made available. Each amendment application made as part of the CLIIP will be processed and noticed in accordance with applicable rules and NRC procedures.

This RIS does not create any new or changed NRC requirements or staff positions, and it requires no specific action or written response. Participation in the CLIIP is purely voluntary.

BACKGROUND INFORMATION

The STS for the five vendor designs include Babcock & Wilcox (NUREG-1430), Westinghouse (NUREG-1431), Combustion Engineering (NUREG-1432), General Electric Boiling Water Reactor/4 (BWR/4) (NUREG-1433), and General Electric BWR/6 (NUREG-1434). The review of a proposed generic change to the STS is a multi-staged process designed to ensure that each STS remains internally consistent, maintains coherence among the various vendors' STS, and incorporates the knowledge and operating experience of the industry and the NRC.

Changes to the STS NUREGs, which are potentially applicable to multiple plants, are proposed to the NRC by the Nuclear Energy Institute (NEI) sponsored Technical Specification Task Force (TSTF) through publicly available submittals. The TSTF includes representatives from the four U.S. commercial nuclear power plant owners groups and NEI. The NRC staff reviews the changes to the STS proposed by the TSTF (referred to as TSTF changes) and will accept, modify, or reject them. Once TSTF changes are accepted, they are considered to be part of the STS. Individual licensees may propose to adopt the TSTF changes during a conversion to the STS or as a separate license amendment application.

The objective of the CLIIP is to improve the efficiencies in the processes for NRC review and licensees' preparation of license amendment applications for NRC-accepted TSTF changes. This is primarily accomplished through multiple licensees being able to use the approved safety evaluation prepared for the TSTF change in connection with amendment applications for specific plants. In an effort to make the NRC work processes more visible, the NRC staff will solicit stakeholder comments on the associated change to the STS, the staff's safety evaluation (SE), and the proposed no significant hazards consideration determination (PNSHCD) before finalizing its acceptance of a TSTF change. Following NRC acceptance of a TSTF change, licensees, as well as the NRC staff, will be able to use the relevant documentation from the NRC-accepted TSTF change in the preparation and processing of license amendment applications. Some of the features of CLIIP incorporate lessons learned from the staff's experiences during the development of the STS and related NRC Policy Statements on TS improvements (e.g., issuing generic letters to announce the availability of "line item improvements" to TS).

The CLIIP would allow efficient adoption of the TSTF changes by licensees that have converted to the STS, as well as by licensees that have not converted to the STS but have determined that the TSTF changes are applicable to their facilities. This process would streamline the documentation process for both the NRC and the licensees. Furthermore, stakeholder involvement would be fostered from the beginning of this process.

SUMMARY OF ISSUE

The purpose of the CLIIP is to streamline the license amendment review process involving TSTF changes applicable to multiple plants. By using a standardized process such as the CLIIP, the burden on an individual licensee would be reduced by saving resources in preparing license amendment applications and, at the same time, the NRC staff review process would become more efficient. The attached flow chart details the process flow for the CLIIP. There are three required participants in the process flow map: the NEI TSTF, the NRC staff, and the licensees. In addition, all NRC stakeholders are provided an opportunity to comment on a proposed TSTF change before NRC acceptance of the change, as well as to participate in the licensing process for each license amendment application. The major aspects of this process are summarized as follows:

1. The CLIIP will improve the efficient adoption of NRC-accepted TSTF changes by having the staff prepare and publish a safety evaluation (SE). A TSTF change request from the NEI TSTF will include a technical justification and a PNSHCD as part of the proposal. The TSTF change process supports subsequent license amendment applications.

2. Following its preliminary review, the NRC staff will use a *Federal Register* notice (FRN) and the NRC website to inform and solicit comments from NRC stakeholders regarding the proposed TSTF changes that will be incorporated into the CLIIP. The stakeholders will be provided with a description of the TSTF change, the staff's preliminary safety evaluation, and a PNSHCD. After the NRC staff resolves the public comments, another FRN and the NRC website will be used to notify NRC stakeholders if the TSTF change has been accepted by the NRC staff and, if accepted, that the TSTF change is available for adoption in proposed plant-specific license amendment applications.

3. The licensees desiring to adopt a specific TSTF change using the CLIIP will need to verify that the proposed change is applicable to their facilities. The NRC announcement and the staff's SE will specify any plant-specific verification or other information required in licensees' applications. The licensees may apply for license amendments by citing the applicability of the PNSHCD and the SE for the accepted TSTF change and addressing any plant-specific information needed to support the staff's review. In order to obtain the maximum efficiency gains from the CLIIP, the NRC will recommend that the licensees submit their applications within a specified time following the FRN announcing that the TSTF change has been accepted.

4. Each amendment application made as part of the CLIIP will be processed and noticed in accordance with applicable rules and NRC procedures. The NRC efficiency gains are achieved by reducing the plant-specific reviews for those changes that are common to multiple licensees.

By using this process, the CLIIP would allow licensees that have converted to the STS to efficiently adopt the accepted TSTF changes subsequent to the conversion. It would also facilitate efficient adoption of accepted TSTF changes as STS evolve for nonconverted plants. Finally, with the licensee's adoption of the uniform description of the proposed change, the PNSHCD, and the SE for a TSTF change request, the CLIIP would provide more disciplined and consistent adoption of the STS by way of a streamlined amendment process.

BACKFIT DISCUSSION

This RIS does not request any action or written response; therefore, the staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment on this RIS was not published in the *Federal Register* because the CLIIP is simply a more effective and efficient application of existing regulations and NRC work processes. The process was developed with opportunities for input from stakeholders during public meetings. The CLIIP adds opportunities for the public to participate in the licensing process.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not request any information collection.

If there is any question about this RIS, please contact the persons listed below.

David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

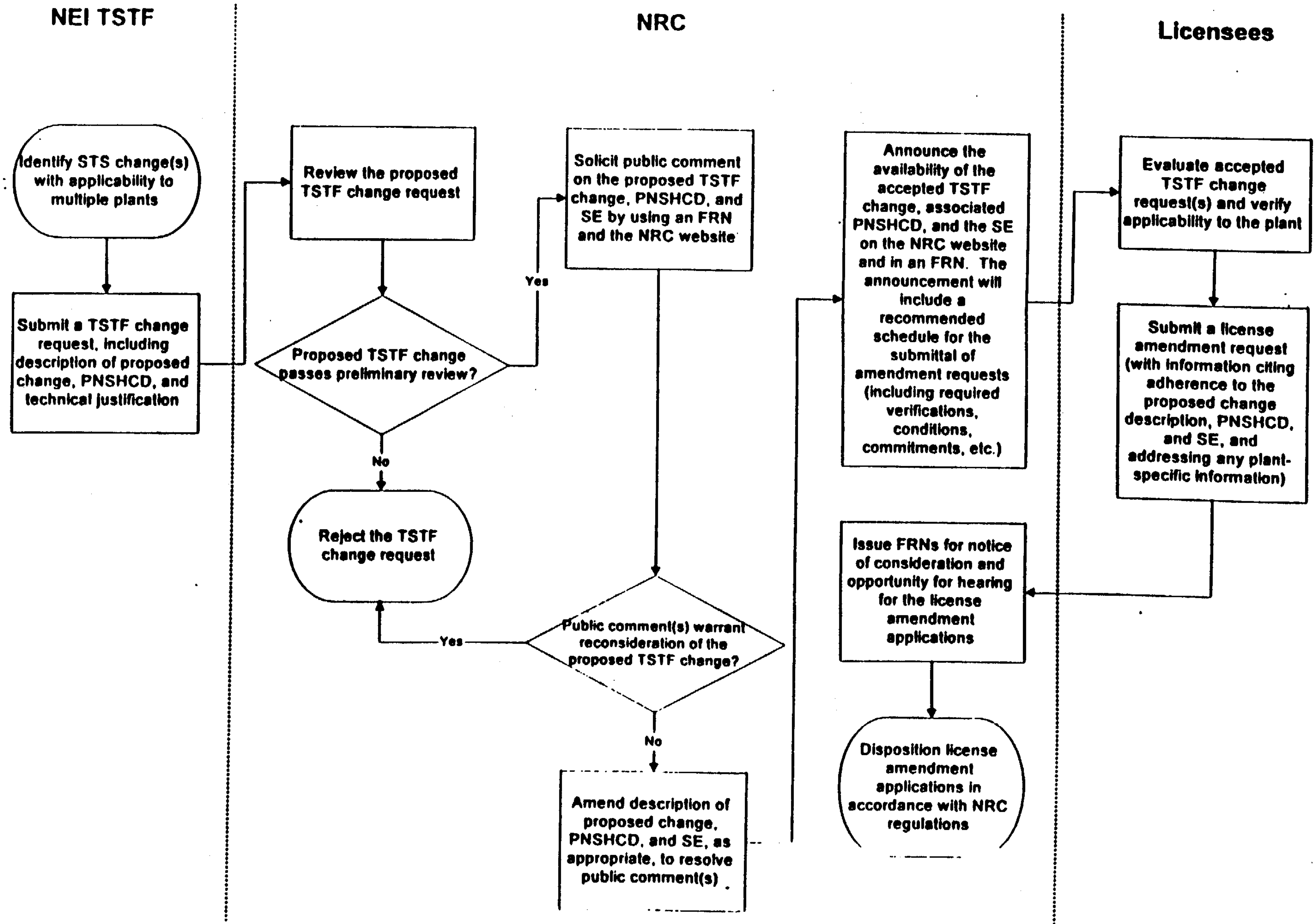
Technical contacts: Tilda Y. Liu, NRR
301-415-1315
E-mail: tyl1@nrc.gov

William D. Reckley, NRR
301-415-1323
E-mail: wdr@nrc.gov

Attachments:
1. Consolidated Line Item Improvement Process (CLIIP) Flow Chart
2. List of Recently Issued NRC Regulatory Issue Summaries

Attachment 1

Consolidated Line Item Improvement Process (CLIIP) Flow Chart



LIST OF RECENTLY ISSUED NRC REGULATORY ISSUE SUMMARIES

Regulatory Issue Summary No.	Subject	Date of Issuance	Issued to
2000-05	Resolution of Generic Safety Issue 165, Spring-Actuated Safety and Relief Valve Reliability	03/16/2000	All holders of OLs for nuclear reactors, except those licensees who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel
2000-04	Operating Reactor Licensing Action Estimates	03/16/2000	All power reactor licensees
2000-03	Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions	03/15/2000	All holders of OLs for nuclear reactors, except for those licensees who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel
2000-02	Closure of Generic Safety Issue 23, Reactor Coolant Pump Seal Failure	02/15/2000	All holders of OLs for nuclear reactors, except for those licensees who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel
2000-01	Changes Concerning Foreign Ownership, Control, or Domination of Nuclear Reactor Licensees	02/01/2000	All holders of OLs for nuclear reactors

OL = Operating Licensing
CP = Construction Permit

STATUS OF VCS PSA PROGRAM

Presented by

Tyndall Estes

V.C. Summer Nuclear Station

May 9, 2000

SCE&G®

A SCANA COMPANY

VCS PSA PREPARATIONS FOR RI ENVIRONMENT

- **IPE CDF** 2.4E-4 TO **5.6E-5**
  - VU / CCW MOD
  - UPDATE AND MODEL REVISIONS
  - REVIEWED ASSUMPTIONS
- **IPEEE FIRE CDF** 4.4E-4 TO **8.5E-5**
  - MODELING DETAILS
  - REVIEWED ASSUMPTIONS

VCS PSA PREPARATIONS FOR RI ENVIRONMENT

- IPE **DAILY USE** ISSUES
  - ADDRESS NRC COMMENTS
  - CAFTA CONVERSION
  - DECREASE QUANTIFICATION RUN TIMES

VCS PSA PREPARATIONS FOR RI ENVIRONMENT

- WESTINGHOUSE RISK-BASED TECHNOLOGY WORKING GROUP
  - **GENERIC ISSUES**
  - PRA QUALITY **STANDARDS**
  - WOG PRA **CERTIFICATION** PROGRAM

VCS PSA PREPARATIONS FOR RI ENVIRONMENT

- EPRI APPLICATION **SOFTWARE USERS GROUP**
  - CAFTA
  - EOOS
- **NRC PRA IMPLEMENTATION PLAN**
  - PARTICIPATED IN NRC TRAINING
  - INDUSTRY DIRECTION GENERALLY SUPPORTS THE NRC PLAN

VCS PSA SUPPORT ITEMS

- **MOV TESTING**
- **APPENDIX J TESTING**
- **RHR LCO EXTENSION**
  - RHR HX MAINTENANCE
- **OSRE PLAN**
  - EQUIPMENT TARGET SETS
- **MAINTENANCE RULE** PROGRAM
  - CONFIGURATION RISK MANAGEMENT PROGRAM REQUIRED IN 2000

VCS PSA SUPPORT ITEMS

- OPERATOR TRAINING RISK SIGNIFICANT INFORMATION
- RHR OUTSIDE ANALYZED CONDITION LER
- RHR COUPLING ON-LINE MODIFICATION
- AOV TESTING PRIORITIZATION

VCS PSA SUPPORT ITEMS

- **EOOS MODELS DEVELOPMENT**
  - OUTAGE REVIEWS
  - MAINT. RULE CONF. RISK MGMT. PROGRAM IN 2000
- VCS **LICENSE RENEWAL** SEVERE ACCIDENT MANAGEMENT ALTERNATIVES REVIEW

CONCLUSIONS

- **RISK INFORMED DECISION MAKING** IS THE NRC's EXPECTATION
- **LIVING PSA PROGRAM** NEEDED FOR RI DECISION MAKING
- MAINT. RULE **CONF. RISK MGMT. PROGRAM** WILL BE REQUIRED
- PSA HAS BECOME A **"COST OF DOING BUSINESS"**

Role of PRA at Virginia Power

NRC Counterparts Meeting

May 9, 2000

Dave Bucheit

Risk Informed Initiatives North Anna (1of 2)

- Allowed Outage Time (AOT)
  - Emergency Diesel
    - 72 hours to 14 days
    - credit for station blackout diesel
  - Pressurizer PORV
    - backup N_2 supplies
    - established 14-day AOT

Risk Informed Initiatives North Anna (2 of 2)

- Surveillance Test Interval (STI) and AOT
  - Reactor protection system
  - Engineered safety features actuation system
  - Summary of changes
    - monthly to quarterly STI
    - outage time for surveillance and maintenance increased (e.g. channel bypass for testing goes from 1 hour to 72 hours)

Risk-Informed Initiatives for Surry

- Risk Informed ISI
  - Unit 1 the pilot plant
    - class I, II and III program
    - risk neutral or lower on system basis
    - substantial reduction in number of inspections
  - Unit 2 submitted Class I program only
    - substantial reduction in number of RCS inspections

Risk Monitoring Tools

- Safety Monitor
  - Solves complete PRA model
  - One minute solution time at 5E-9 truncation
  - Dual unit, all modes model
  - Station user interface to select equipment to be removed from service
- ORAM shutdown assessment

Initiatives In Process

- Dropped Rod Time Calculation
  - Eliminate seismic term
  - RG 1.174 approach
- Physical Security Plan Review
- Support WOG RBTWG
  - RG 1.174 justification for various TS AOT
  - Risk Informed Technical Specifications (RITS)

10CFR50.65(a)(4)

- Rely on Safety Monitor for Modes 1-4
- Use ORAM for Modes 5-6
- Modify document management information system (DMIS) to enhance risk awareness throughout the station
- Enhance existing procedures
- Concern about getting it right

Significance Determination Process

- Recent events have required use of SDP
  - NAPS trip from mode 4
  - Surry 58 fan verbatim compliance
- Comments
  - PRA group involved quickly
  - SDP worksheets not always helpful
    - shutdown
    - fire
  - Need prompt communication
    - with station personnel
    - with senior reactor analyst in region

SCE&G
A SCANA COMPANY

South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, SC 29065
(803) 345-5209
(803) 635-1461

Gary J. Taylor
Vice President
Nuclear Operations

July 1, 1998
RC-98-0126

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Mark Padovan

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
TECHNICAL SPECIFICATION CHANGE - TSP 980004
SNUBBERS

South Carolina Electric and Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests an amendment to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) in accordance with 10CFR50.90. This proposed amendment will revise the VCSNS TS 4.7.7.e to remove the "situational" surveillance requirement of "during shutdown" following the specified surveillance interval of "at least once per eighteen months". Also, administrative changes are proposed to Surveillance Requirement 4.7.7.g and to BASES 3/4.2.2 and 3/4.2.3 to correct typographical errors.

Removal of the situational requirement, "during shutdown", will allow VCSNS to accomplish TS surveillance testing of snubbers on-line. In addition, TS 4.0.2 would be applicable to the "specified surveillance interval".

The amendment request is contained in the following documents:

Attachment I	Explanation of Changes Summary Marked-up Technical Specification Pages Revised Technical Specification Pages
Attachment II	Safety Evaluation
Attachment III	No Significant Hazards Determination
Attachment IV	Environmental Impact Determination

This proposed TS amendment request has been reviewed by both the Plant Safety Review Committee and the Nuclear Safety Review Committee.
Document Control Desk

SCE&G requests approval of this change to the VCSNS TS by January 31, 1999. This will allow implementation within 60 days prior to refueling outage 11.

I declare that these statements and matters set forth herein are true and correct to the best of my knowledge, information and belief.

Should you have questions, please call Mr. Jim Turkett at (803) 345-4047.

Very Truly Yours,

David Lavigne for

Gary J. Taylor

JT/GJT/jt
Attachments (4)

c. J.L. Skolds
W.F. Conway
R.R. Mahan (w/o Attachments)
R.J. White
L.A. Reyes
NRC Resident Inspector
M.K. Batavia
J.B. Knotts, Jr.
P. Ledbetter
RTS (TSP 980004)
File (813.20)
DMS (RC-98-0126)

STATE OF SOUTH CAROLINA :
: TO WIT :
COUNTY OF FAIRFIELD :

I hereby certify that on the 1st day of July 1998, before me, the subscriber, a Notary Public of the State of South Carolina personally appeared David A. Lavigne, being duly sworn, and states that he has signature authority for the Vice President, Nuclear Operations of the South Carolina Electric & Gas Company, a corporation of the State of South Carolina, that he provides the foregoing response for the purposes therein set forth, that the statements made are true and correct to the best of his knowledge, information, and belief, and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal

Michael J Zaccone
Notary Public

My Commission Expires

Date

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SCE&G – EXPLANATION OF CHANGES

Page	Affected Section	Bar #	Description of Change	Reason for Change
3/4 7-17	4.7.7.e	1	Removed Situational qualifier "during shutdown" from frequency statement.	Follows safety considerations presented by GL 91-04 which states "The staff concludes that TS need not restrict surveillances as only being performed during shutdown." with proper regard for the effect of on-line surveillances on plant safety.
3/4 7-19	4.7.7.g	1	~~2nd~~ 3rd paragraph - Change "4.7.6.e" to "4.7.7.e".	Typo - Administrative correction to reflect proper reference.
B 3/4 2-3	BASES 3/4.2.2 and 3/4.2.3	1	2nd paragraph, subparagraph a. - Change "$\pm$ 13 steps" to "$\pm$ 12 steps"	Typo - Administrative correction to reflect proper rod steps utilized by Westinghouse in original design analysis. Reconciles Bases to LCO 3/4.1.3.1., Action Statement b. and d.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

(i) manually induced snubber movement; (ii) evaluation of in-place snubber piston setting; or (iii) stroking the mechanical snubber through its full range of travel.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubbers to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (i) the cause for being classified as unacceptable is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (ii) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.7.f. When a fluid port of a hydraulic snubber is found to be uncovered the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing unless the test is started with the piston in the as found setting, extending the piston rod in the tension mode direction. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable and may be reclassified as acceptable for determining the next inspection interval provided that criterion (i) and (ii) above are met. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements of 3.7.7 shall be met.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter ~~during shutdown~~, a representative sample of either: (1) At least 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.7.f, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested, or (2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1, "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.7.f. The cumulative number of snubbers of a type tested is denoted by "N." At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Accept" region, testing of that type of snubber may be terminated.

g. Functional Test Failure Analysis (Continued)

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.6.e for snubbers not meeting the functional test acceptance criteria. 7

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the seals service life is not exceeded between surveillance inspections. The maximum expected service life for the various seals, seal materials, and applications shall be determined and established based on engineering information and the seals shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

POWER DISTRIBUTION LIMIT

BASES

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± ~~13~~ 12 steps, indicated, from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through c. above are maintained. As noted on the RCS Total Flow Rate Versus R figure in the CORE OPERATING LIMITS REPORT (COLR), RCS flow rate and power may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if core power is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated in 3.2.3 and used in the RCS Total Flow Rate Versus R figure in the COLR, accounts for $F^N_{\Delta H}$ less than or equal to the $F^{RTP}_{\Delta H}$ limit specified in the COLR. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature and thus is the maximum "as measured" value allowed.

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset any rod bow penalty and transition core penalty. The remaining margin is available for plant design flexibility.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

SAFETY EVALUATION
FOR REVISING THE SNUBBER FUNCTIONAL TESTING FREQUENCY
IN THE VIRGIL C. SUMMER NUCLEAR STATION
TECHNICAL SPECIFICATIONS

DESCRIPTION OF AMENDMENT REQUEST

This license amendment request proposes to revise Surveillance Requirement 4.7.7.e to remove the "situational" qualifying condition of "during shutdown" from the specified test interval. This change is in conformance with recommendations and guidance presented in Generic Letter 91-04, Enclosure 1, industry experience, and the forthcoming issuance of ASME OM Code , Subsection ISTD. An administrative change is proposed to Surveillance Requirement 4.7.7.g and to BASES 3/4.2.2 and 3/4.2.3 to correct typographical errors.

SAFETY EVALUATION

Snubber installation, removal, repair, and functional testing at VCSNS is procedurally implemented and controlled. Removal for maintenance or testing is subject to engineering evaluation prior to unpinning. The evaluations consider system availability, plant configuration, and current operational mode. The surveillance practice implemented at VCSNS usually allows only one snubber on a piping train to be removed at a time. This practice precludes significant dynamic effects on the associated piping system should an event occur while the snubber is unpinned.

This process is within general industry practice and is supported by the results of NUREG/CR-6027 (EGG-2697), *Preliminary Evaluation of Snubber Single Failures, April 1993*. This NUREG utilized snubber history experience gained from utilities of varying reactor types to analyze the sensitivity of a system response to a single inoperable snubber resulting from an event outside the bounds of the analyses performed by the licensees to establish the licensing basis of their plants. The NUREG presented in the Conclusion section that only the PWR ice condenser main steam line penetration was judged to be potentially vulnerable to the failure of a single snubber under blowdown and/or seismic loads. Also the NUREG notes, "The piping systems assessed can withstand several times the design safe shutdown earthquake before rupture, based on recent tests of piping systems that have shown them to have significant reserve safety margin when subjected to earthquake loads.". The Conclusion ended with the statement, "The overall risk from the potential single failure of most piping snubbers appears to be low.".

Removal of the situational requirement, "during shutdown", will allow VCSNS to accomplish TS surveillance testing on-line. In addition, TS 4.0.2 would be applicable to the "specified surveillance interval".

VCSNS TS Surveillance Requirement 4.7.7.e denotes snubber functional testing to be performed "at least once per 18 months during shutdown". Industry experience since this TS was developed has led to an ongoing evaluation and restructuring of the standards regulating snubber examinations and testing. The NRC has actively participated with utility groups, standards committees and manufacturers throughout the evolution of snubber surveillance development. The issue of on-line examination and testing will be addressed by the ASME OM Code in *Code for Operation and Maintenance of Nuclear Power Plants, Subsection ISTD, Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light Water Reactor Power Plants.*

On-line examination and testing of snubbers was ISTD Working Group Action Item 96-1. This item was approved by the Main Committee and by the Board on Nuclear Codes and Standards. The change to allow on-line snubber testing will be published in the 1998 edition of the ASME OM Code.

Additionally, the NRC has issued interpretations and Generic Letters in regards to TS surveillance requirements which recognize industry experience and has modified surveillance requirements which presented hardships or unnecessary restrictions on utilities with no corresponding benefit to nuclear safety. This has resulted in recommendations for improving TS through allowable changes or by providing guidelines to adopt industry practices.

The NRC (*USNRC - Technical Specifications Interpretations; Fermi, 5/18/88*) noted that a "regular surveillance interval" is an interval "characterized by the wording 'at least once per' a specified time interval. A 'situational' surveillance requirement is characterized by the wording 'within' a specified time interval and is followed by a certain condition or situation (i.e., prior to startup, after control rod movement, after taking a sample, etc.)." As TS 4.7.7.e utilizes both a "regular surveillance interval" ("at least once per 18 months") and a "situational" ("during shutdown") surveillance in the same sentence; it would seem prudent, based on the technical justification described herein, to delete the "situational" requirement.

Also, Generic Letter 91-04, (*CHANGES IN TECHNICAL SPECIFICATION SURVEILLANCE INTERVALS TO ACCOMMODATE A 24-MONTH FUEL CYCLE, 4/2/91*) notes that "the added restriction to perform certain surveillances during shutdown may be misinterpreted". It also states "This restriction ensures that a surveillance would only be performed when it is consistent with safe plant operation.". Enclosure 1 recommends that licensees desiring to adopt a 24 month fuel cycle submit a TS change to define the nominal frequency for surveillances that are specified to be performed each refueling interval. The NRC additionally states that licensees may omit the qualification of "during shutdown". Even though this generic letter was primarily issued to address adoption of a 24 month fuel cycle, the safety considerations discussed for adhering to surveillance activities are applicable for any length fuel cycle and should be permitted for allowing on-line snubber surveillance testing.

In addition, the NRC has approved at least one TS change allowing on-line testing of snubbers (reference TAC #M92804, March 4, 1996).

CONCLUSION

On-line performance of surveillance testing has been demonstrated through industry experience and NRC evaluations (e.g., GL 91-04) to be appropriate and acceptable with proper consideration to plant safety and public risk. In particular, NUREG/CR-6027 allows that the risk for loss of a single snubber (e.g., on-line snubber testing) is low.

Pursuant to the preceding information, the proposed TS amendment request does not create any potential degradation in the ability of the associated piping systems to perform their design functions under postulated environmental or seismic conditions.

NO SIGNIFICANT HAZARDS DETERMINATION FOR REVISING THE SNUBBER FUNCTIONAL TESTING FREQUENCY IN THE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

DESCRIPTION OF AMENDMENT REQUEST

This license amendment request proposes to revise Surveillance Requirement 4.7.7.e to remove the "situational" qualifying condition of "during shutdown" from the specified test interval. This change is in conformance with recommendations and guidance presented in Generic Letter 91-04, Enclosure 1, industry experience, and the forthcoming issuance of ASME OM Code , Subsection ISTD. An administrative change is proposed to Surveillance Requirement 4.7.7.g and to BASES 3/4.2.2 and 3/4.2.3 to correct typographical errors.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

SCE&G has evaluated the proposed changes to the VCSNS TS described above against the Significant Hazards Criteria of 10 CFR 50.92 and determined that the changes do not involve any significant hazard for the following reasons:

1. The probability or consequences of an accident previously evaluated is not significantly increased.

   The proposed change will not affect system operation or performance, nor do they affect any Engineered Safety Features actuation setpoints or accident mitigation capabilities. NUREG/CR-6027 supports the determination that piping failure due to a snubber single failure is considered low. Therefore, the proposed changes will not significantly increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

2. The possibility of an accident or a malfunction of a different type than any previously evaluated is not created.

   The changes to the situational testing requirements will not affect the method of operation of any system to which a snubber is attached. The proposed changes only address the plant mode at which a surveillance activity may be performed. No new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of these changes. Therefore, the possibility of a new or different kind of accident other than those already evaluated will not be created by this change.

3. The margin of safety has not been significantly reduced.

   This proposed change will not have an impact on the overall reliability of the snubber population. This is due, in part, to the fact that the snubber test plans are self correcting. As functional test failures are identified, additional snubbers are required to be tested. Thus, the reliability of the snubber population is maintained. The proposed change does not alter the intent or method by which the surveillances are conducted, does not involve any physical changes to the plant, does not alter the way any structure, system, or component functions, and does not modify the manner in which the plant is operated. Therefore the proposed change will not degrade the ability of the snubbers to perform their safety function or significantly decrease the margin of safety.

Based on the above discussions, it has been determined that the requested technical specification changes do not involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; nor create the possibility of a new or different kind of accident or condition over previous evaluations; nor involve a significant reduction in a margin of safety. Therefore, pursuant to 10 CFR 50.91, the preceding analysis provides a determination that the proposed TS amendment request poses no significant hazard as delineated by 10 CFR 50.92.

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ENVIRONMENTAL IMPACT DETERMINATION FOR REVISING THE SNUBBER FUNCTIONAL TESTING FREQUENCY IN THE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as specified below:

1. The amendment involves no significant hazards determination.

   As demonstrated in Attachment IV, the proposed change does not involve any significant hazards consideration.

2. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

   The proposed changes do not involve a change to the facility or operating procedures which would create new types of effluents. The change to allow on-line snubber testing will not affect system performance or operation. This change will not compromise the recognized effluents. The limits of 10 CFR 100 and 10 CFR 50 Appendix A, GDC 19 are not impacted.

3. There is no significant increase in individual or cumulative occupation radiation exposure.

   The proposed change will not create a significant increase in radiation exposure due to the required surveillance activity, rather, performing on-line surveillances on accessible systems should result in lower exposures than if the surveillance were to be deferred until shutdown when certain systems generate more radioactivity and result in potentially more radiation exposure risk to plant personnel.

Based on the above, it is concluded that there will be no impact on the environment resulting from the proposed changes and that the proposed changes meet the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to requiring a specific environmental assessment by the Commission.

LIC

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

April 28, 1999

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 99-128
NL&OS/GDM R0
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGE
REFUELING WATER CHEMICAL ADDITION TANK MINIMUM VOLUME

Pursuant to 10CFR50.90, Virginia Electric and Power Company requests amendments, in the form of revisions to the Technical Specifications to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2. The proposed change will reduce the minimum volume requirement for the refueling water chemical addition tank to permit implementation of a more reasonable level setpoint for maintaining tank level. Also incorporated in this submittal is a minor administrative revision to a Technical Specification table. A discussion of the proposed Technical Specifications change is provided in Attachment 1.

The proposed Technical Specifications change has been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee. It has been determined that the proposed Technical Specifications change does not involve an unreviewed safety question, as defined in 10CFR50.59. Marked-up Technical Specifications pages that reflect the proposed change are provided in Attachment 2. Revised Technical Specifications pages that incorporate the proposed change are provided in Attachment 3. The basis for our determination that the Technical Specifications change does not involve a significant hazards, as defined in 10CFR50.92, is provided in Attachment 4.

Should you have any questions or require additional information, please contact us.

Very truly yours,

D. A. Christian
Vice President – Nuclear Operations

Attachments:
1. Discussion of Change
2. Mark-up of Technical Specifications
3. Proposed Technical Specifications
4. Significant Hazards Consideration Determination

Commitments made in this letter: None.

cc: U.S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth Street, SW
Suite 23T85
Atlanta, Georgia 30303

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

Commissioner
Department of Radiological Health
Room 104A
1500 East Main Street
Richmond, VA 23219

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by D. A. Christian, who is Vice President - Nuclear Operations, for J. P. O'Hanlon, who is Senior Vice President - Nuclear, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 28th day of April, 1999.

My Commission Expires: March 31, 2000.

Maggie McClure
Notary Public

(SEAL)

ATTACHMENT 1

DISCUSSION OF CHANGE

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2

DISCUSSION OF CHANGE

Introduction

Virginia Electric and Power Company is proposing a revision to the Surry Power Station Technical Specification 3.4.A.4 to reduce the refueling water chemical addition tank (CAT) minimum volume requirement. As part of an overall engineering review of Technical Specifications (TS) setting limits, the minimum volume requirement for the refueling water CAT was re-evaluated. As a result of this re-evaluation, it was determined that the minimum refueling water CAT volume could be decreased to provide additional operating margin. The revised TS minimum CAT volume requirement reflects the re-evaluated CAT volume assumed in the accident analyses plus instrument uncertainties, as well as additional positive margin for conservatism.

No increase in the probability of occurrence or consequences of an accident or equipment malfunction will result from the proposed TS change. The revised minimum TS volume requirement only affects the required CAT volume used to mitigate accidents, and the revised minimum TS refueling water CAT volume continues to ensure that the available CAT volume is sufficient to meet accident analyses assumptions for accident mitigation. Implementing the proposed change does not create the possibility of an accident of a different type than was previously evaluated in the safety analysis report, since the minimum volume requirement for the CAT does not introduce any new accident precursors or modes of operation. The proposed change continues to ensure that accident analyses assumptions are maintained. Although the minimum CAT volume is being decreased, the revised limit continues to ensure that the post-LOCA containment spray and containment sump pH, and post-LOCA recirculation switchover are acceptable. Therefore, the margin of safety as defined in the TS bases is unaffected.

Minor administrative changes are also being implemented in TS Table 4.1-2B by this Technical Specifications change.

Background

As part of an overall engineering review of TS setting limits, the minimum volume requirement for the refueling water CAT was re-evaluated. The current Technical Specifications volume and concentration requirements for the refueling water CAT (i.e., 4200 gallons of 17% to 18 % NaOH) ensure that the containment spray pH during a Loss of Coolant Accident is maintained above 8.5 and the containment sump pH remains above 7.0. These pH levels are necessary to ensure that: 1) the decontamination factors assumed in the accident analyses for elemental and particulate iodine removal coefficients are maintained, and 2) the potential for stress corrosion cracking is minimized.

Operation at the current TS limit of 4200 gallons is difficult, since the TS limit is near the volume capacity of the CAT (i.e. 4330 gallons). Consequently, when considering instrument error, only a limited operating margin is available. Therefore, a re-evaluation of the safety analysis limit for the minimum CAT volume was performed to determine whether a lower minimum CAT volume would be acceptable, thus permitting the CAT volume TS limit to be lowered. A lower TS limit would provide greater operational flexibility in maintaining the required CAT volume, including instrument error, and preclude inadvertently overflowing of the tank. The proposed TS minimum CAT volume also permits establishing an alarm setpoint above the proposed TS limit that will provide additional operational flexibility in maintaining the required tank volume.

Licensing Basis

- Minimum CAT volume change

The minimum refueling water CAT volume specified in the original TS 3.4.A.4 was 3,360 gallons of solution with a sodium hydroxide concentration of 18 percent by weight. This value corresponded to the RWST minimum volume requirement to ensure that a sufficient amount of caustic was added to the containment spray system relative to the water provided by the RWST. This volumetric relationship ensured accident analysis assumptions for containment spray system performance were adequately met.

In 1980 and 1981, the TS were revised for Surry Unit 2 and Unit 1 via Amendment Nos. 59 and 71, respectively, to increase the required minimum volume for the CAT from 3360 gallons to 4200 gallons, and to establish a required concentration range for sodium hydroxide of not less than 17 percent and not greater than 18 percent. The increase in the CAT minimum volume was required to compensate for an increase in the minimum volume requirement for the Refueling Water Storage Tank due to containment spray system modifications.

A minor revision was made to the text of TS 3.4.A.4 in TS Amendment Nos. 180 and 180 for Surry Units 1 and 2 to make the wording more consistent in terminology and format.

- Administrative Changes

Minor administrative changes are also being implemented by this TS change request. In a letter dated January 30, 1996 (Serial No. 96-005), Virginia Electric and Power Company requested changes to the TS to eliminate the surveillance requirement for certain reactor coolant liquid samples under particular specified conditions. These sampling changes were approved in TS Amendments 209 for Surry Units 1 and 2, and incorporated into TS Table 4.1-2B, Minimum Frequencies for Sampling Tests. However, separate from the requested TS changes, the test requirements for Item 6, Secondary Coolant, of the Table were inadvertently altered in format such that they no longer aligned with the testing frequencies specified for that item. Consequently, the text in the "Test" section of Item 6 in Table 4.1-2B has been returned to its correct format to properly align with the corresponding test frequencies in the next column. The symbols used for beta and gamma have also been spelled out for greater clarity, and the FSAR reference has been deleted as UFSAR section 10.3 provides no pertinent reference information for this item. The FSAR section reference header for the Table has also been revised to read UFSAR rather than FSAR to indicate that the Updated Final Safety Analysis Report is the accurate reference source.

Design Basis

A refueling water CAT is located near the RWST for each unit. The CAT stores sodium hydroxide solution that is added to the containment spray system water from the RWST by balanced gravity feed directly to the suction of the containment spray pumps. The level in the CAT is designed to follow level in the RWST as the two tanks empty together. This ensures that the pressure head of each tank remains in the same proportion at the suction of the containment spray pumps, and that the concentration of the sodium hydroxide being injected into containment remains constant. The sodium hydroxide is injected into the containment via the containment spray system to enhance iodine removal from the containment atmosphere and to control sump water pH by keeping it slightly basic. A specific range of pH is required for effective removal of volatile iodine species from the containment atmosphere and retention in the containment sump water, and also serves as a preventative against chloride stress corrosion.

The CAT is a vertical, cylindrical tank with a capacity of 4330 gallons. Technical Specifications require that the CAT contain at least 4200 gallons of 17 to 18 % sodium hydroxide during normal plant operation. To address instrument uncertainties and the change in specific gravity of the liquid in the CAT caused by the difference in sodium hydroxide concentration, CAT level is currently not allowed to decrease below 98.4% during normal operation to ensure that the minimum TS volume is not violated.

Discussion

The impact of the reduction in the safety analysis limit minimum CAT volume to 3800 gallons was evaluated to determine an acceptable TS setpoint limit for minimum CAT volume. It was determined that the proposed change potentially affected three aspects of the Surry accident analyses:

1) The containment analysis (e.g., the CAT is a source of relatively cold water for injection into containment).

2) The containment spray and post-LOCA sump pH (e.g., a reduction in the quantity of the NaOH solution from the CAT could reduce the pH of the containment spray and the liquid in the sump, thereby diminishing the capacity of

the spray and sump liquid to remove and retain volatile iodine species from the containment atmosphere).

3) The post-LOCA recirculation switchover time (e.g., a reduction in the quantity of NaOH solution from the CAT could reduce the time to boric acid precipitation in the core region following a large break LOCA, necessitating a reduction in the post-LOCA cold-to-hot leg recirculation switchover interval presented in the Emergency Operating Procedures.)

It was determined that the proposed reduction in the safety analysis limit for CAT inventory from 4200 gallons to 3800 gallons did not significantly impact the results of the containment analysis, the containment spray and post-LOCA sump pH analysis, or the post-LOCA recirculation switchover time analysis. Specifically:

- The CAT inventory is not credited as a source of energy removal in the containment analysis of record.

- Reducing the assumed CAT volume from 4200 gallons to 3800 gallons results in a small reduction in the calculated minimum sump and spray pH, but the calculated pH values remain above the minimum sump and spray pH acceptance criteria (7.0 and 8.5, respectively), consistent with NUREG-0800, Standard Review Plan, Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, December 1988.

- The CAT volume is not considered in the analysis, which supports the currently applicable EOP recirculation switchover interval.

Therefore, an acceptable reduction in the safety analysis limit for the CAT inventory to 3800 gallons permits establishing a revised TS minimum CAT volume of 3930 gallons. This value corresponds to a 91.0% tank level (for 18% sodium hydroxide concentration at 68°F). The 3800 gallon safety analysis limit CAT volume corresponds to an 87.9% tank level. The difference between this level (i.e., 87.9% level) and the level associated with the proposed TS minimum CAT volume (i.e., 91.0% level) is 3.1%, which is greater than the Channel Statistical Allowance (CSA) of 2.6% associated with the CAT level indication and the emergency response facility computer system (ERFCS) computer point. The additional 0.5% level margin between the current licensing basis safety analysis limit CAT volume (3800 gallons, or 87.9% level) and the proposed TS minimum CAT volume (3930 gallons, or 91.0% level) remains available to compensate for potential future changes in the calculated CAT level CSA.

Specific Changes

- Technical Specification 3.4.A.4 is revised as follows to reflect the reduced refueling water CAT minimum volume requirement:

  4. The refueling water chemical addition tank shall contain at least 3930 gallons of solution with a sodium hydroxide concentration of at least 17 percent by weight but not greater than 18 percent by weight.

- Technical Specification Table 4.1-2B, Minimum Frequencies for Sampling Tests, Item 6, Secondary Coolant, is revised as follows to 1) indicate that two separate tests are required with different test frequencies, 2) replace the symbols used for beta and gamma activity with the actual words for clarity and 3) delete the inappropriate FSAR reference:

6. Secondary Coolant	Fifteen minute degassed beta and gamma activity	Once/72 hours
	DOSE EQUIVALENT I-131	Monthly(4) Semiannually (8)

The FSAR section reference header for Table 4.1-2B has also been revised to read UFSAR rather than FSAR to indicate that the Updated Final Safety Analysis Report is the accurate reference source.

Safety Significance

This change to the TS reduces the minimum volume requirement for the refueling water CAT from 4200 to 3930 gallons. The proposed change continues to provide assurance to assure that accident analyses assumptions will remain valid, and that the effect of instrument uncertainties during accident conditions are adequately addressed. The changes to TS Table 4.1-2B are strictly administrative in nature.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated are not increased. When the revised safety analysis limit minimum CAT volume of 3800 gallons was implemented, consideration was given to the effects of the proposed reduced CAT volume on containment integrity analyses, containment spray and post-LOCA sump pH analyses, and the post-LOCA recirculation switchover time interval specified in Emergency Operating Procedures. The reduced value was determined to be acceptable. The proposed TS minimum CAT volume (3930 gallons) includes an allowance for the CAT level Channel Statistical Allowance (CSA), so that the safety analysis limit CAT volume (3800 gallons) will not be violated when the measured CAT volume (i.e., tank level) is above the TS minimum CAT volume. Because the affected accident analyses have been evaluated and found to meet their acceptance criteria with the reduced safety analysis limit CAT volume, the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased. The proposed reduction in the TS minimum CAT volume has no bearing on the probability of occurrence of any accident previously evaluated, since neither the volume of the CAT nor the sodium hydroxide inventory in the CAT have any bearing on postulated accident initiators.

The possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created. The proposed reduction in the TS minimum CAT volume does not involve any alterations to the physical plant which introduce any new or unique operational modes or accident precursors. The proposed TS minimum CAT volume permits establishing an alarm setpoint above the proposed TS limit that will provide additional operational flexibility in maintaining the required tank volume.

The margin of safety as defined in the basis of the Technical Specifications is not reduced. It was determined that the affected safety analyses continue to meet their respective acceptance criteria with the revised minimum CAT volume assumption. By implementing the proposed change in the TS minimum CAT volume, a CAT level alarm setpoint may be established which includes a conservative allowance for level measurement uncertainty to ensure that neither the TS limit nor the safety analysis limit for minimum CAT volume will be violated at the time a CAT level alarm is received. Therefore, it is concluded that the proposed change is being made to provide greater assurance that the margin of safety defined in the TS bases is maintained.

ATTACHMENT 2

MARK-UP OF TECHNICAL SPECIFICATIONS

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2

3.4 SPRAY SYSTEMS

Applicability

Applies to the operational status of the Spray Systems.

Objective

To define those limiting conditions for operation of the Spray Systems necessary to assure safe unit operation.

Specification

A. A unit's Reactor Coolant System temperature or pressure shall not be made to exceed 350°F or 450 psig, respectively, unless the following Spray System conditions in the unit are met:

1. Two Containment Spray Subsystems, including containment spray pumps, piping, and valves shall be OPERABLE.

2. Four Recirculation Spray Subsystems, including recirculation spray pumps, coolers, piping, and valves shall be OPERABLE.

3. The refueling water storage tank shall contain at least 387,100 gallons of borated water at a maximum temperature of 45°F. The boron concentration shall be at least 2300 ppm but not greater than 2500 ppm.

4. The refueling water chemical addition tank shall contain at least 3930 ~~4,200~~ gallons of solution with a sodium hydroxide concentration of at least 17 percent by weight but not greater than 18 percent by weight.

5. All valves, piping, and interlocks associated with the above components which are required to operate under accident conditions shall be OPERABLE.

TABLE 4.1-2B
MINIMUM FREQUENCIES FOR SAMPLING TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
1.	Reactor Coolant Liquid Samples	Radio-Chemical Analysis(1)	Monthly(5)	
		Gross Activity(2)	5 days/week(5)	9.1
		Tritium Activity	Weekly (5)	9.1
		* Chemistry (CL, F & O_2)	5 days/week(9)	4
		* Boron Concentration	Twice/week	9.1
		$\bar{E}$ Determination	Semiannually(3)	
		DOSE EQUIVALENT I-131	Once/2 weeks(5)	
		Radio-iodine Analysis (including I-131, I-133 & I-135)	Once/4 hours(6) and (7) below	
2	Refueling Water Storage	Chemistry (Cl & F)	Weekly	6
3	Boric Acid Tanks	* Boron Concentration	Twice/Week	9.1
4	Chemical Additive Tank	NaOH Concentration	Monthly	6
5	Spent Fuel Pit	* Boron Concentration	Monthly	9.5
6	Secondary Coolant	~~Fifteen minute degassed β and γ activity~~ beta gamma DOSE EQUIVALENT I-131	Once/72 hours Monthly(4) Semiannually(8)	~~10.3~~
7	Stack Gas Iodine and Particulate Samples	* I-131 and particulate radioactive releases	Weekly	

* See Specification 4.1.D

(1) A radiochemical analysis will be made to evaluate the following corrosion products: Cr-51, Fe-59, Mn-54, Co-58, and Co-60.

(2) A gross beta-gamma degassed activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of µCi/cc.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2

3.4 SPRAY SYSTEMS

Applicability

Applies to the operational status of the Spray Systems.

Objective

To define those limiting conditions for operation of the Spray Systems necessary to assure safe unit operation.

Specification

A. A unit's Reactor Coolant System temperature or pressure shall not be made to exceed 350°F or 450 psig, respectively, unless the following Spray System conditions in the unit are met:

1. Two Containment Spray Subsystems, including containment spray pumps, piping, and valves shall be OPERABLE.

2. Four Recirculation Spray Subsystems, including recirculation spray pumps, coolers, piping, and valves shall be OPERABLE.

3. The refueling water storage tank shall contain at least 387,100 gallons of borated water at a maximum temperature of 45°F. The boron concentration shall be at least 2300 ppm but not greater than 2500 ppm.

4. The refueling water chemical addition tank shall contain at least 3930 gallons of solution with a sodium hydroxide concentration of at least 17 percent by weight but not greater than 18 percent by weight.

5. All valves, piping, and interlocks associated with the above components which are required to operate under accident conditions shall be OPERABLE.

TABLE 4.1-2B
MINIMUM FREQUENCIES FOR SAMPLING TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
1.	Reactor Coolant Liquid Samples	Radio-Chemical Analysis(1)	Monthly(5)	
		Gross Activity(2)	5 days/week(5)	9.1
		Tritium Activity	Weekly (5)	9.1
		* Chemistry (CL, F & O_2)	5 days/week(9)	4
		* Boron Concentration	Twice/week	9.1
		$\bar{E}$ Determination	Semiannually(3)	
		DOSE EQUIVALENT I-131	Once/2 weeks(5)	
		Radio-iodine Analysis (including I-131, I-133 & I-135)	Once/4 hours(6) and (7) below	
2.	Refueling Water Storage	Chemistry (Cl & F)	Weekly	6
3.	Boric Acid Tanks	* Boron Concentration	Twice/Week	9.1
4.	Chemical Additive Tank	NaOH Concentration	Monthly	6
5.	Spent Fuel Pit	* Boron Concentration	Monthly	9.5
6.	Secondary Coolant	Fifteen minute degassed beta and gamma activity	Once/72 hours	
		DOSE EQUIVALENT I-131	Monthly(4) Semiannually(8)	
7.	Stack Gas Iodine and Particulate Samples	* I-131 and particulate radioactive releases	Weekly	

* See Specification 4.1.D

(1) A radiochemical analysis will be made to evaluate the following corrosion products: Cr-51, Fe-59, Mn-54, Co-58, and Co-60.

(2) A gross beta-gamma degassed activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc.

ATTACHMENT 4

SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2

Significant Hazards Consideration

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed Technical Specifications (TS) change for Surry Power Station Units 1 and 2 and determined that a significant hazards consideration does not exist. The proposed change will reduce the TS minimum volume requirement for the refueling water chemical addition tank (CAT) from a setting limit of 4200 gallons to 3930 gallons. Although the minimum CAT volume is being decreased, the revised limit continues to ensure that the post-LOCA containment spray, containment sump pH, and post-LOCA recirculation switchover are acceptable, and accident analyses assumptions are maintained. The remaining TS changes to Item 6 in TS Table 4.1-2B is strictly administrative in nature. The basis for this determination is provided as follows:

Criterion 1 - <u>Does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

The probability or the consequences of an accident previously evaluated are not increased. When the revised Safety Analysis Limit minimum CAT volume of 3800 gallons was implemented, consideration was given to the effects of the proposed reduced CAT volume on containment integrity analyses, containment spray and post-LOCA sump pH analyses, and the post-LOCA recirculation switchover time interval specified in Emergency Operating Procedures. The change was determined to be acceptable as accident analyses assumptions would continue to be met. The proposed TS minimum CAT volume (3930 gallons) includes an allowance for the CAT level Channel Statistical Allowance (CSA), so that the safety analysis limit CAT volume (3800 gallons) will not be violated when the measured CAT volume (i.e., tank level) is at or above the TS minimum CAT volume limit. The proposed reduction in the TS minimum CAT volume has no bearing on the probability of occurrence of any accident previously evaluated, since neither the volume nor the sodium hydroxide inventory of the CAT have any bearing on postulated accident initiators. Furthermore, because the affected accident analyses have been evaluated and found to meet their acceptance criteria with the reduced safety analysis limit CAT volume, the consequences of an accident previously evaluated is not increased.

Criterion 2 - <u>Does not create the possibility of a new or different kind of accident from any accident previously evaluated</u>.

The possibility of a new or different kind of accident than any accident previously evaluated is not created. The proposed reduction in the TS minimum CAT volume does not involve any alterations to the physical plant that would introduce any new or unique operational modes or accident precursors. Only the TS minimum CAT volume is being changed to establish an operationally feasible alarm setpoint to provide the operators additional flexibility in maintaining the required CAT volume.

Criterion 3 - Does not involve a significant reduction in a margin of safety.

The margin of safety is not reduced. It was determined that the affected safety analyses continue to meet their respective acceptance criteria with the revised minimum CAT volume. By implementing the proposed change in the TS minimum CAT volume, a CAT level alarm setpoint may be established which includes a conservative allowance for level measurement uncertainty such that neither the proposed TS minimum CAT volume nor the Safety Analysis Limit CAT volume will be violated at the time a CAT level alarm is received. Therefore, it is concluded that the proposed change will not reduce the margin of safety.

This analysis demonstrates that the proposed amendment to the Surry Units 1 and 2 Technical Specifications does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident and does not involve a significant reduction in a margin of safety.

LIC

99-581

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 1, 1999

Mr. J. P. O'Hanlon
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: MINIMUM VOLUME OF SODIUM HYDROXIDE SOLUTION IN CHEMICAL ADDITION TANK (TAC NOS. MA5470 AND MA5471)

Dear Mr. O'Hanlon:

The Commission has issued the enclosed Amendment No. 222 to Facility Operating License No. DPR-32 and Amendment No. 222 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TS) in response to your application transmitted by letter dated April 28, 1999.

These amendments revise TS Section 3.4.A.4 for Units 1 and 2. The changes relax the minimum volume requirement for the refueling water Chemical Addition Tank (CAT) from 4200 gallons to 3930 gallons. The CAT provides sodium hydroxide solution which is mixed with water from the Refueling Water Storage Tank in the event of an accident. The resulting solution is then fed to the suction of containment spray pumps. The change will provide additional operating flexibility while maintaining the proper pH in the containment spray solution and the containment sump.

A minor administrative change is also being made to TS Table 4.1-2B to correct an earlier printing error and to delete a reference which no longer applies.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

G E Edison

Gordon E. Edison, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 222 to DPR-32
2. Amendment No. 222 to DPR-37
3. Safety Evaluation

cc w/encls: See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 222
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated April 28, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 222 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 1, 1999

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 222
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated April 28, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 222 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 1, 1999

ATTACHMENT TO

LICENSE AMENDMENT NO. 222 TO FACILITY OPERATING LICENSE NO. DPR-32

LICENSE AMENDMENT NO. 222 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Remove Page	Insert Page
TS 3.4-1	TS 3.4-1
TS 4.1-10	TS 4.1-10

3.4 SPRAY SYSTEMS

Applicability

Applies to the operational status of the Spray Systems.

Objective

To define those limiting conditions for operation of the Spray Systems necessary to assure safe unit operation.

Specification

A. A unit's Reactor Coolant System temperature or pressure shall not be made to exceed 350°F or 450 psig, respectively, unless the following Spray System conditions in the unit are met:

1. Two Containment Spray Subsystems, including containment spray pumps, piping, and valves shall be OPERABLE.

2. Four Recirculation Spray Subsystems, including recirculation spray pumps, coolers, piping, and valves shall be OPERABLE.

3. The refueling water storage tank shall contain at least 387,100 gallons of borated water at a maximum temperature of 45°F. The boron concentration shall be at least 2300 ppm but not greater than 2500 ppm.

4. The refueling water chemical addition tank shall contain at least 3930 gallons of solution with a sodium hydroxide concentration of at least 17 percent by weight but not greater than 18 percent by weight.

5. All valves, piping, and interlocks associated with the above components which are required to operate under accident conditions shall be OPERABLE.

TABLE 4.1-2B
MINIMUM FREQUENCIES FOR SAMPLING TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
1.	Reactor Coolant Liquid Samples	Radio-Chemical Analysis(1)	Monthly(5)	
		Gross Activity(2)	5 days/week(5)	9.1
		Tritium Activity	Weekly (5)	9.1
		* Chemistry (CL, F & O_2)	5 days/week(9)	4
		* Boron Concentration	Twice/week	9.1
		$\bar{E}$ Determination	Semiannually(3)	
		DOSE EQUIVALENT I-131	Once/2 weeks(5)	
		Radio-iodine Analysis (including I-131, I-133 & I-135)	Once/4 hours(6) and (7) below	
2.	Refueling Water Storage	Chemistry (Cl & F)	Weekly	6
3.	Boric Acid Tanks	* Boron Concentration	Twice/Week	9.1
4.	Chemical Additive Tank	NaOH Concentration	Monthly	6
5.	Spent Fuel Pit	* Boron Concentration	Monthly	9.5
6.	Secondary Coolant	Fifteen minute degassed beta and gamma activity	Once/72 hours	
		DOSE EQUIVALENT I-131	Monthly(4) Semiannually(8)	
7	Stack Gas Iodine and Particulate Samples	* I-131 and particulate radioactive releases	Weekly	

* See Specification 4.1.D

(1) A radiochemical analysis will be made to evaluate the following corrosion products: Cr-51, Fe-59, Mn-54, Co-58, and Co-60.

(2) A gross beta-gamma degassed activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 222 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated April 28, 1999, Virginia Electric and Power Company (the licensee) proposed amendments to the Technical Specifications (TS) for Surry Power Station, Units 1 and 2. The proposed amendments would decrease the minimum-required volume of sodium hydroxide solution in the chemical addition tank (CAT) from 4200 gallons to 3930 gallons. The licensee requested this change in order to provide additional operating margin for the CAT. Engineering evaluation performed by the licensee has indicated that the decreased volume in the CAT will still provide enough sodium hydroxide to the water coming from the Refueling Water Storage Tank (RWST) and other sources of borated water to maintain pH of the spray solution and the containment sump at specified values.

The proposed amendment includes a minor administrative change to Table 4.1-2B in the TS which specifies minimum frequencies for different sampling tests. These changes consist of slightly modifying the format of the table and clarifications.

2.0 EVALUATION

The CAT contains sodium hydroxide solution which is gravity-fed to the borated water coming from the RWST in order to maintain alkaline pH in the post-LOCA sprays and containment sump. The current TS requires a minimum of 4200 gallons of sodium hydroxide solution at between 17 and 18 percent concentration in order to ensure that the pH of the spray solution and containment sump is maintained at or above 8.5 and 7.0, respectively. Maintaining these pH values is needed to ensure that no stress corrosion cracking or reevolution of radioactive iodine will take place in the post-LOCA environment. However, 4200 gallons of solution in the CAT, which has a capacity of 4330 gallons, provides a very narrow operational margin. In order to increase this margin, the licensee reevaluated the minimum volume of sodium hydroxide solution needed for maintaining the required alkalinity levels. The results of this evaluation have indicated that reducing the volume of 17 to 18 percent sodium hydroxide solution to 3800 gallons causes only a very slight decrease in pH which never goes below the specified limits. This allowed the licensee to specify for the CAT a minimum volume of 3930 gallons of sodium hydroxide solution. This volume includes a margin of 3.1 percent which is greater than the instrument channel statistical allowance associated with the CAT level indication.

The staff has reviewed the assumptions and methodologies used by the licensee in its analyses to justify the requested modifications. The staff also performed an independent verification of the licensee's calculations. The staff found that all the justifications were well supported by the appropriate licensee analyses.

The proposed change to Table 4.1-2B is acceptable because there is no safety impact of correcting an earlier misprint and deleting a reference that is no longer applicable.

The staff has reviewed the modification to the Surry Power Station, Units 1 and 2 TS proposed by the licensee. The proposed modification changes the minimum required volume of sodium hydroxide solution in the CAT. Based on its review, the staff concludes that all the TS changes proposed in this submittal are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comment.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 48869). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, 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 Contributor: K. Parczewski

Date: November 1, 1999

LICENSING WORKSHOP

UFSAR UPDATING

STEPHEN R. MONARQUE
PROJECT MANAGER
DIVISION OF LICENSING PROJECT MANAGEMENT
OFFICE OF NUCLEAR REACTOR REGULATION

May 10, 2000

UFSAR OVERVIEW

- 10 CFR 50.71(e)
  - LICENSEES REQUIRED TO PERIODICALLY UPDATE THE UFSAR TO REFLECT THE CURRENT DESIGN AND OPERATION OF THE FACILITY

- NUCLEAR ENERGY INSTITUTE (NEI) 98-03, Revision 1

- NRC REGULATORY GUIDE 1.181

UFSAR UPDATE

- NEW REQUIREMENTS
- FACILITY OR PROCEDURE CHANGES
- NEW SAFETY ISSUES ANALYSES
- LEVEL OF DETAIL

NEW REQUIREMENTS

- NEW OR MODIFIED DESIGN BASES

- SUMMARY OF NEW OR MODIFIED SAFETY ANALYSES

- APPROPRIATE UFSAR DESCRIPTION

FACILITY OR PROCEDURE CHANGES

- CHANGES TO DESIGN BASES, SAFETY ANALYSES, OR CHANGES TO DESCRIPTIONS OF EXISTING STRUCTURES, SYSTEMS, OR FUNCTIONS IN UFSAR

- CHANGE RESULTS IN THE REMOVAL OF SSCs DESCRIBED IN THE UFSAR OR ELIMINATION OF FUNCTIONS OR PROCEDURES IN UFSAR

FACILITY OR PROCEDURE CHANGES (Continued)

- CHANGE OR SUPPORTING SAFETY EVALUATION RESULTS IN NEW DESIGN OR SAFETY ANALYSES IN UFSAR

NEW SAFETY ISSUES ANALYSES

- LICENSEES SHOULD EVALUATE THE EFFECTS OF ANALYSES IN RESPONSE TO NRC REQUESTS, GENERIC LETTERS, OR BULLETINS

- INCLUDE IN UFSAR UPDATES, IF EXISTING DESIGN BASES, SAFETY ANALYSES, OR UFSAR DESCRIPTION ARE NOT ACCURATE

LEVEL OF DETAIL

- IS THE UPDATED INFORMATION SUFFICIENT TO PERMIT UNDERSTANDING OF NEW OR MODIFIED SAFETY ANALYSES, DESIGN BASES AND FACILITY OPERATION

CLARIFYING THE DEFINITION OF DESIGN BASES

KAREN COTTON
NRC

OBJECTIVE

THE NRC HAS ISSUED A DRAFT GUIDE FOR COMMENT TITLED, "GUIDANCE AND EXAMPLES FOR IDENTIFYING 10CFR 50.2 DESIGN BASES."

THE DRAFT GUIDANCE WAS DEVELOPED TO PROVIDE A CLEARED UNDERSTANDING OF WHAT CONSTITUTES DESIGN BASES INFORMATION AS DEFINED IN 10 CFR 50.2

10 CFR 50.2 DEFINITION

DESIGN BASES means

- That information which identifies the specific function to be performed by a structure, system or component of a facility
- The specific values or ranges of values chosen for controlling parameters as refence bounds for design.

10 CFR 50.2 DEFINITION cont.

These Values may be

- Restraints derived from gennerally accepted "state of the art" practices for achieving funcional goals
- Requirements derived from analysis of effects of a postulated accident for which a structure, system or component must meet its functional goals

Relevance of Design Bases

- Design Bases" used in following regulations:
  - 50.4 (FSAR content)
  - 50.59 (Changes)
  - 50.72,50.73 (Reporting)
  - Appendix A to Part 50 (GDC)
  - Appendix B to Part 50 (QA)
- Useful for evaluating degraded and conforming conditions

History

- Engineering team inspections (Late 1980s)
- Industry guidance (NUMARC 90 12)
- NUREG-1397 (February 1991)
- Commission Policy Statement (August 1992)
- Millstone/Maine Yankee (1996)
- Nine Mile Point - reporting issue (1997)
- Revised industry guidance (NEI 97-04)
- Staff committed to develop regulatory guidance

General Guidance

- Design bases functions:
  - Functions performed by SSCs that are
    - Required to meet regulations, license conditions, orders or technical specifications
    - Credited in safety analysis to meet NRC requirements
- Design bases values:
  - Values or ranges of values of controlling parameters established by NRC requirenent
    - Established or confirmed by safety analyses
    - Chosen by the licensee from an applicable code,standard or guidance document as refence bounds for design to meet design bases functional requirements

Summary

- The Draft Guide proposes endorsement to Nuclear Energy Institutes document Appendix B, Guidlines and Examples for Identifying 10 CFR 50.2 , Design Bases," to NEI 97 - 04, Design Bases Program Guidlines."
- The Guide is published in the federal register for public comment. The comment period ends June 15, 2000.

10 CFR 50.72/50.73 Proposed Rulemaking

- Karen Cotton
- NRR

Objective

- To present/discuss the status of changes to regulated reporting requirements in 10 CFR 50.72 (Immediate Notification Requirements) and 50.73 (Licensee Event Report System)

Topics of Discussion

- Background
- Objectives of Rulemaking
- Significant Changes
- Schedule

Background

- Advanced Notice of Proposed Rulemaking (ANPR) issued on 7/23/98
- Public Meetings held on 8/21/98, 9/1/98 and 11/13/98
- Proposed Rule published on 7/6/99
- Public meetings held 8/3/99, 2/25/00, and 3/22/00.
- Final rule provided to the Commission 4/21/00

OBJECTIVES

- To better align reporting requirements with the NRC's reporting needs
- To reduce the reporting burden, consistent with NRC's reporting needs
- To clarify the reporting requirements where needed
- To maintain consistency with NRC actions to improve integrated plant assessments

Principal Changes

- Required Reporting Times
- Late Surveillance Tests
- Reporting of Historical Problems
- Outside the Design Basis of the Plant
- ESF Actuation

Schedule

- After Commission Approval, the Final Rule will be provided to OMB for review and clearance under the paperwork reduction act.
- About 3 months later, the Final Rule will be published

NRC RESOURCES

- Dennis Allison
  - Generic Issues, Environmental, Financial and Rulemaking Branch
  - (301) 415-1178
  - dpa@nrc.gov
- Internet
  - link on NRC website (www.nrc.gov)
  - ruleforum.llnl.gov or
- SECY-00-0093, when released, provides details of the Final Rule recommended to the Commission

10 CFR 50.59 Rulemaking

Rich Emch
NRC/VEPCO/SCANA
Licensing Workshop

Schedule

- 10/4/99 - Final Rule published
- 2/22/00 - Revised NEI 96-07 submitted
- 4/25/00 - Draft Reg Guide published
- 6/9/00 - Public comment period closes
- 7/00 - Final version of NEI 96-07 due
- 9/30/00 - Final Reg Guide due to Commission
- - Rule effective 90 days after final Reg Guide is published

Major Changes

- Removal of reference to "Unreviewed Safety Question"
- Term "Safety Evaluation" is changed to "10 CFR 50.59 Evaluation"
- Added definitions of "change" and "facility as defined in final safety analysis (as updated)

Major Changes (continued)

- Will allow for "minimal" changes without requiring prior NRC approval
- Changed "probability" statement to "increase in frequency" or "likelihood of occurrence"
- "Malfunction of a different type" is being replaced with "malfunction with a different result"

Major Changes (continued)

- Margin of Safety Evaluation is being replaced with 2 new criteria
- Criterion (vii) - Evaluation of integrity of fission product barriers
- Criterion (viii) - Changes to approved evaluation methods

Impacts & Benefits

- Impacts:
  - Will require major revision to 50.59 procedure
  - Will require new training standards to be developed
- Benefits:
  - Overall improvement over previous rule language
  - Agreed upon industry/NRC guidance

Issues to Be Resolved

- Screening on "Affects Design Function"
- NEI and NRC agree in principle
- NRC concerned that resultant screening may not be broad enough

Submitting Relief Requests to the NRC

Dr. G. E. Edison, NRC Senior Project Manager

10 CFR 50.55a Subjects

Subjects	**10 CFR 50.55a Paragraph**
Reactor Coolant Pressure[1] Boundary *Sect. III - Class 1 Components*	50.55a(c)
Quality Group B Components[1,2] *Sect. III - Class 2 Components*	50.55a(d)
Quality Group C Components[1,2] *Sect. III - Class 3 Components*	50.55a(e)
Inservice Testing Items *Sect. XI - Class 1,2,3,4*	50.55a(f)
Inservice Inspection (examination) Items *Sext. XI - Class 1,2,3,MC,CC*	50.55a(g)
Protection Systems *IEEE-279*	50.55a(h)

<u>Notes</u>: 1. *Applies to Design*
2. *For CP after 1984 - Not applicable to USA plants*

Methods to Use to Ask for Relief

I. Propose an alternative to the code requirement and show that:

- the alternative provides an <u>acceptable level of quality and safety</u> pursuant to **10 CFR 50.55a(a)(3)(i)**, or
- complying with the code requirement would result in <u>hardship or unusual difficulty</u> (excessive cost and time) without a compensating increase in quality or safety pursuant to **10 CFR 50.55a(a)(3)(ii).**

II. Show that the code requirement is <u>impractical</u> (impossible - not just inconvenient) pursuant to **10 CFR 50.55a(f)(6)(i)** <u>for inservice testing</u> items or **50.55a(g)(6)(i)** <u>for inservice inspection (examination)</u> items.

III. Use of a later ASME Code Edition pursuant to **10 CFR 50.55a(f)(4)(iv)** <u>for inservice testing</u> items or **50.55a(g)(4)(iv)** <u>for inservice inspection (examination)</u> items.

<u>Note:</u> *Applies for Code Edition endorsed by staff. Current approved Code - 1995 Edition. Staff has not yet approved the 1998 Edition.*

Methods the NRC Can Use to Authorize an Alternative or Grant Relief

- Authorize a licensee-proposed alternative in accordance with **10 CFR 50.55a(a)(3)(i)** if NRC determines that the alternative provides an acceptable level of quality and safety, or

- Authorize a licensee-proposed alternative (if any) in accordance with **10 CFR 50.55a(a)(3)(ii)** if NRC determines that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, or

- Grant relief and impose alternative requirements in accordance with **10 CFR 50.55a(f)(6)(i)** for inservice testing items if NRC determines that the code requirement is impractical, or

- Grant relief and impose alternative requirements in accordance with **10 CFR 50.55a(g)(6)(i)** for inservice inspection (examination) items if NRC determines that the code requirement is impractical.

- Approve request for Alternative by use of staff endorsed later ASME Code Edition (currently 1995 Edition) in accordance with **10 CFR 50.55a(f)(4)(iv)** for inservice testing items or **50.55a(g)(4)(iv)** for inservice inspection (examination) items.

Table 1 — Relief Requests Detailed Guidance

10 CFR 50.55a Section	**Applicable Table**
10 CFR 50.55a**(a)(3)(i)**	**see Table 2**
10 CFR 50.55a**(a)(3)(ii)**	**see Table 3**
10 CFR 50.55a**(f)(6)(i)**	**see Table 4**
10 CFR 50.55a**(g)(6)(i)**	**see Table 5**
10 CFR 50.55a**(g)(6)(ii)**	**see Table 5**
10 CFR 50.55a**(f)(4)(iv)** 10 CFR 50.55a**(g)(4)(iv)**	**see Table 6**

☞ Note: Pick the single, most applicable 10 CFR 50.55a section to address.

☞ Note: The NRC can only authorize an alternative that the utility proposes in their written submittal. The utility must prepare another written submittal proposing (other) alternatives if they decide or agree with the NRC to use (other) alternatives.

☞ Note: 64FR51370 addresses Code Cases N-513 & N-523-1 Flaw Repair of Class 2 and Class 3 Piping.

Table 2 — Authorizing a Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i)

Purpose	<u>Authorize</u> a utility-proposed alternative in accordance with **10 CFR 50.55a(a)(3)(i).**
Necessary Determination	Determine if the utility-proposed alternative provides an <u>acceptable level of quality and safety</u>.
Guidance	➻ Indicate the applicable Code edition and addenda, and describe the Code requirement.
	➻ Describe the proposed alternative <u>and bases</u>.
	➻ Discuss why the proposed alternative provides an acceptable level of quality and safety.
	➻ Specify the duration and scope of the proposed alternative.
	➻ Do not mention impracticality, burden, unusual difficulty or hardship.

Table 3 Authorizing a Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(ii)

Purpose	Authorize a utility's proposed alternative in accordance with **10 CFR 50.55a(a)(3)(ii).**
Necessary Determinations	Determine if complying with the specified requirement would result in hardship or unusual difficulty (rather than being impractical) without a compensating increase in the level of quality and safety.
	For ISI items — Determine if the proposed alternative provides reasonable assurance of pressure boundary integrity.
	For IST items — Determine if the proposed alternative provides reasonable assurance that the component or system is operationally ready (capable of performing its intended function).
Guidance	➼ Indicate the applicable Code edition and addenda, and describe the Code requirement.
	➼ Describe the utility-proposed alternative and bases.
	➼ Discuss why complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.
	➼ For IST items: Discuss why the proposed alternative provides reasonable assurance that the component or system is operationally ready.
	➼ For ISI items: Discuss why the proposed alternative provides reasonable assurance of pressure boundary integrity.
	➼ Specify the duration and scope of the proposed alternative.
	➼ Do not mention impracticality.

Table 4 Inservice Testing — Granting Relief in Accordance with 10 CFR 50.55a(f)(6)(i)

Purpose	Grant relief and impose alternative requirements in accordance with **10 CFR 50.55a(f)(6)(i)** for inservice testing items.
Necessary Determinations	Determine if the code requirement is impractical.
	Determine if the proposed testing provides reasonable assurance that the component is operationally ready (capable of performing its intended function).
Guidance	➻ Indicate the applicable Code edition and addenda.
	➻ Describe the utility's proposed alternative (if any) and bases.
	➻ Describe why it is impractical for the utility to comply with the specified requirement.
	➻ Describe the burden on the utility created by imposing the requirement (e.g., having to replace a component, redesign the system or shutdown the plant).
	➻ Discuss why the proposed testing provides reasonable assurance that the component is operationally ready.
	☞ Note: **10 CFR 50.55a(f)(6)(i)** allows the NRC to impose additional requirements without having the utility first commit to them. **10 CFR 50.55a(a)(3)** does not allow this.
	➻ Specify the duration and scope of the alternative.
	➻ Do not mention hardship or unusual difficulty.

Table 5 Inservice Inspection — Granting Relief in Accordance with 10 CFR 50.55a(g)(6)(i)

Purpose	Grant relief and impose alternative requirements in accordance with **10 CFR 50.55a(g)(6)(i)** for inservice inspection (examination).
Necessary Determinations	Determine if the code requirement is impractical.
	Determine if the proposed inservice inspection (examination) provides reasonable assurance of component or structure pressure boundary integrity.
Guidance	➳ Additional guidance in Generic Letter 90-05 "Guidance for Performing Temporary Non-code Repair of ASME Code Class 1, 2, and 3 Piping."
	➳ Indicate the applicable Code edition and addenda, and describe the Code requirement.
	➳ Describe the proposed alternative (if any) and bases.
	➳ Describe why it is impractical to comply with the specified requirement.
	➳ Describe the burden created by imposing the requirement (e.g., having to replace a component, redesign the system or shutdown the plant).
	➳ Describe why the proposed inspection (examination) provides reasonable assurance of component or structure pressure boundary integrity.
	☞ Note: **10 CFR 50.55a(g)(6)(i)** allows the NRC to impose additional requirements without having the utility first commit to them. **10 CFR 50.55a(a)(3)** does not allow this.
	➳ Specify the duration and scope of the alternative.
	➳ Do not mention hardship or unusual difficulty.

Table 6— Approving Use of Later ASME Code Edition-1995 Edition 10 CFR 50.55a(f)(4)(iv) and 10CFR 50.55a(g)(4)(iv)

Purpose	Approve utility proposed alternative in accordance with **10 CFR 50.55a(f)(4)(iv)** for inservice testing items or **50.55a(g)(4)(iv)** for inservice inspection (examination) items.
Necessary Determination	Determine if the utility-proposed alternative relates to portions of the 1995 Edition, while meeting other related requirements.
Guidance	➻ Indicate the applicable Code edition and addenda, and describe the Code requirement.
	➻ Describe the proposed alternative.
	➻ Discuss why the proposed alternative meets the approved Code Edition.
	➻ Specify the duration and scope of the proposed alternative.
	➻ Do not mention impracticality, burden, unusual difficulty or hardship.

CRITERIA FOR ENVIRONMENTAL ASSESSMENTS

Dr. G. E. Edison

May 10, 2000

NEPA

(National Environmental Policy Act—1969)

- All Federal Agencies Must Comply
- NRC Regulations in 10 CFR 51

NRC REGULATIONS IMPLEMENTING NEPA

10 CFR 51.20.....ENVIRONMENTAL IMPACT STATEMENT (EIS)

10 CFR 51.21.....ENVIRONMENTAL ASSESSMENT (EA)

10 CFR 51.22.....CATEGORICAL EXCLUSIONS

10 CFR 51.41.....INFORMATION SUBMITTALS

10 CFR 51.45.....ENVIRONMENTAL REPORTS(ERs)

10 CFR 51.53(c).....POST-CONSTRUCTION ERs

PROPOSALS REQUIRING ENVIRONMENTAL REPORTS

LICENSE RENEWAL......10 CFR 51.53(c)

LICENSE TO MAUFACTURE.....10 CFR 51.54
A NUCLEAR REACTOR

MATERIALS LICENSES......10 CFR 51.60

INDEPENDENT SPENT FUEL......10 CFR 51.61
STORAGE INSTALLATION(ISFSI)

LAND DISPOSAL OF...... 10 CFR 51.62
RADIOACTIVE WASTE

CONTENT OF ENVIRONMENTAL REPORTS (10 CFR 51.45)

IMPACT ON ENVIRONMENT

ADVERSE ENVIRONMENTAL EFFECTS

ALTERNATIVES TO PROPOSED ACTION

LOCAL SHORT TERM USE OF ENVIRONMENT VS LONG TERM

IRREVERSIBLE AND IRRETRIEVABLE COMMITMENTS OF RESOURCES

ANALYSIS OF ENVIRONMENTAL IMPACTS VS ALTERNATIVES TO PROPOSAL AND TO ADVERSE EFFECTS, COSTS VS BENEFITS

COMPLIANCE WITH OTHER PERMITS AND REGULATIONS

MEETING 10 CFR 51.21(EAs) FOR CPs AND ORs

NUREG 1555............STANDARD REVIEW PLAN
FOR ERs

OFFICE LETTER 906.......STAFF GUIDANCE
FOR EAs

MEETING 10 CFR 51.53(c)(ERs) FOR LICENSE RENEWAL

REGULATORY GUIDE 4.2, SUPPLEMENT 1 (TO BE PUBLISHED) PROVIDES GUIDANCE AND FORMAT FOR ERs

NUREG 1437 IS A GENERIC ENVIRONMENTAL IMPACT STATEMENT(EIS)

NUREG 1437 SUPPLEMENTS ARE PLANT-SPECIFIC EISs

NUREG 1555, SUPPLEMENT1, IS STAFF'S STANDARD REVIEW PLAN FOR ERs