June 14, 2000

LICENSEE: FirstEnergy Nuclear Operating Company

FACILITIES: Beaver Valley Nuclear Power Station, Units 1 and 2
Davis-Besse Nuclear Power Station
Perry Nuclear Power Plant

SUBJECT: NRC/FENOC LICENSING WORKSHOP MEETING SUMMARY, MAY 24 - 25, 2000

The Nuclear Regulatory Commission (NRC) and the FirstEnergy Nuclear Operating Company (FENOC) jointly sponsored a licensing workshop on May 24 and 25, 2000, in Akron, Ohio. Attendees included staff of FENOC, the NRC, and a representative from both the Nuclear Energy Institute (NEI) and the State of Pennsylvania. The goals of the workshop were to (1) promote an understanding of the licensing process, (2) improve licensing submittal quality, (3) enhance the regulatory interface, and (4) provide information on current regulatory issues.

Representatives from the Office of Nuclear Reactor Regulation (NRR) presented information on the role of the NRR project manager, regulatory processes, risk informed licensing actions, electronic information exchange, and reporting requirements. NEI presented the status of activities in the licensing action task force. The workshop also included a breakout session to discuss licensing submittal quality. A copy of the workshop book is included as Enclosure 1. Enclosure 2 is a summary of feedback received from the workshop attendees. Enclosure 3 is a list of the workshop attendees.

/RA/
Douglas V. Pickett, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-334, 50-346, 50-412, 50-440

Enclosures:
1. Workshop handouts
2. Feedback
3. Attendees

cc w/encls: See next page

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Acceptance Review

- Oath & affirmation, State copy
- Clear description of change
- Safety analysis and justification
- NSHC and EA (or exclusion)
- Approval and implementation schedules
- Is it risk-informed?

Work Planning

- PM (and Technical Staff)
  - Search for precedents
  - Review method (PM, tech staff, etc.)
  - Scope & depth of review
  - Resource planning and schedule
  - Priority

Priorities

- Priority 1
  - Highly risk-significant safety concern
  - Issue involving plant shutdown, derate, or restart
  - Compliance with statutory requirements

Priorities (continued)

- Priority 2
  - Significant safety issue
  - Support continued safe plant operations
  - Determine significance of operating event
  - Risk-informed licensing action
  - Topical report with near-term or significant safety benefit

Priorities(continued)

- Priority 3
  - Moderate to low safety significance
  - Cost beneficial licensing actions
  - Generic issue or multi-plant action
  - Topical report with limited benefit

NSHC Determination

- NSHC Based on 50.92 (51 FR 7751)
  - Significant increase in probability or consequences of an accident
  - Possible new or different accident
  - Significant reduction in margin of safety
- If proposed NSHC, hearing <u>can</u> be after amendment
- If SHC or no determination, any hearing would precede amendment

Environmental Assessments

- Environmental Impact Statements (EISs) and EAs based on 51.20 to 51.22
  - EISs very rare
  - Amendment EA exclusions in 51.22
  - Most amendments meet the exclusions
  - EA must be published in the Federal Register before the amendment is issued

Noticing

- "Normal" amendments, 50.91(a)(2)
  - Bi-weekly or individual *Federal Register* notices - 30 day comment period
  - Notice of proposed amendment, proposed NSHC, hearing opportunity
  - Notice of issuance
- If a proposed NSHC determination is not made, use individual notices
  - Can't be handled as an exigent or emergency

Noticing - Exigent amendment

- Notice in Federal Register (FR) if amendment is to be issued after 15 days but before 30 days
  - Individual FR notice
  - Repeat in bi-weekly FR notice
- Notice in local media if amendment is to be issued after 6 days but before 15 days
  - Repeat in bi-weekly FR notice
- Amendments require a final NSHC determination

Noticing - Emergency Amendment

- Emergency amendments noticed after issuance for comment and an opportunity for hearing

Reviewer Assignments

- Reviews can be performed by PM or technical staff, considerations include:
  - Technical complexity & risk significance
  - PM technical expertise
  - Conformance to improved Standard Technical Specifications (ISTS) guidance
  - Conformance to precedents
  - Resource availability & schedule needs

Review Process And Documents Preparation

- Review process
  - Precedents
  - Requests for additional information (RAIs)
  - Regulatory commitments
- Document preparation
  - Safety evaluation
  - Concurrence review
  - Amendment issuance

Review Process And Documents Preparation

- Precedents
  - Ensure request meets current expectations
    - Format
    - Guidance to industry
    - Technical content

Review Process And Documents Preparation

- Requests for additional information
  - Staff goal: 1 RAI per reviewing technical branch
  - Early communication with licensee
    - Resolve minor issues
    - Clarify questions
    - Establish reasonable response date

Commitments

- Regulatory commitments are information relied on by the staff in making its conclusion but are not included in the technical specifications.

- Current staff practice outlined in SECY-98-224, NRC Guidance on Commitment Management

Commitments

- Hierarchy of licensing-basis infcrmation
  - ✓ Obligations - license, TS, rules, orders
  - ✓ Mandated Licensing-Basis Information - UFSAR, QA/security/emergency plans
  - ✓ Regulatory Commitments - docketed statements agreeing or volunteering to take specific actions
  - ✓ Non-Licensing-Basis Information

Commitments

- Commitments stated in the safety evaluation are considered part of the licensing basis but are not legally binding requirements
- Safety evaluation should clearly state what actions are considered regulatory commitments
- Control of commitments is in accordance with licensees' programs

Commitments

- Escalation to license conditions reserved for safety-significant matters (e.g., those that meet 10 CFR 50.36 criteria for inclusion)

- Staff is continuing to include license conditions for relocation of information to USAR or other controlled documents in amendment implementation condition

Commitments

- Office Letter 900 to be issued spring 2000
  - Will address NEI's revised guidance
  - Will include "audits" of licensee's Commitment Management Program
    - ✓ performed by PMs
    - ✓ 1/3 of plants per year

Safety Evaluation

- Routinely included
  - Staff evaluation - why the request satisfies regulatory requirements
  - State consultation
  - Environmental considerations
- As needed
  - Regulatory commitments
  - Emergency/exigent provisions
  - Final NSHC determination

Concurrence

- Licensing Assistant
  - format and revised TS pages
- Technical Branch
  - technical adequacy
- Technical Specifications Branch
  - Significant deviations from iSTS guidance or changes consistent with iSTS
  - Use of 10 CFR 50.36 criteria
- Office of the General Counsel
  - Legal defensibility and completeness

Amendment Issuance

- Ensure that we've addressed all comments from public and state
- Transmitted to licensee via letter
  - Issued <u>after</u> associated EA
- Standard distribution (cc) list
  - Notify NRC staff of licensee's organization changes to list via docketed letter
- ***Federal Register*** notice of issuance

References

- NRR Office Letter 803, Rev. 3
- 10 CFR 50.30 (Applications)
- 10 CFR 50.90 (Amendment Applications)
- 10 CFR 50.91 (Noticing, State Consult.)
- 10 CFR 2.105 (Noticing)
- 10 CFR 50.92 (NSHC, Issuance)
- 10 CFR 51.20-22 (EIS and EA)
- 10 CFR 50.36 (TS Criteria)
- SECY 98-244 (Commitments)

Regulatory Processes

Dan Collins, Project Manager
Project Directorate 1
Division of Licensing Project Management

Introduction

- Goals
  - Educate
  - Develop Ideas for Improvement both at NRC and Utilities
  - Stimulate Discussion for Breakout Sessions
- Discuss Processes for Change
  - Licensee Controlled
  - NRC Controlled
- Provide Overview of Each Change Process

License Amendment - 10 CFR 50.90

- Requirements
  - Submit as specified in 10 CFR 50.4
  - Fully describe changes; follow form of original application
  - No significant hazards consideration [50.92(c)]
    - » No significant increase in probability or consequences of an accident previously evaluated
    - » No possibility of a new/different kind of accident from any previously evaluated
    - » No significant reduction in margin of safety

License Amendment - 10 CFR 50.90 - continued

- Content
  - Oath and affirmation
  - Description of amendment
  - Deterministic safety assessment
  - Optional - supported by risk-informed information
  - No significant hazards consideration
  - Environment input
    - » To support impact statement per 10 CFR 51.20
    - » To support environmental assessment per 10 CFR 51.21
    - » None if exclusion applies per 10 CFR 51.22(c)
  - Revised Technical Specifications or License Condition

License Amendment - 10 CFR 50.90 - continued

- Content (con't)
  - New or revised commitments identified
  - New or revised Design Basis (10 CFR 50.2) and Licensing Basis identified
  - Reference to current licensing basis
  - Cost Beneficial Licensing Actions (NRC AL95-02)
    - » Total lifetime savings identified
  - Need date and basis identified
  - Implementation schedule provided

Request to Modify License (2.206)

- Criteria
  - Generally meant for public use and imposing civil penalties
  - Specify action requested and set forth facts
  - If not submitted by licensee, any licensee input at NRR request or by 50.54(f)
  - Licensee may be party to any hearing
- Content
  - Petition by Licensee
    - » Safety analysis to support action requested
    - » Set forth facts
    - » Specify Tech Specs, license conditions to be modified/added
    - » Environmental Analysis
    - » Information to initiate hearing/support subsequent Order

Relief Requests: 10 CFR 50.55a

- Criteria
  - Alternatives - would provide acceptable level of quality and safety
  - Hardship or unusual difficulty - without compensating increase in level of quality or safety
  - Impractical - design, materials, access limitations [IST: 50.55a(f)(6)(i); ISI: 50.55a(g)(6)(i)]
  - Augmented - may be required, in conjunction with "impractical" relief if:
    - » Added assurance of operational readiness is needed (IST)
    - » Added assurance of structural reliability is needed (ISI)

Relief Requests: 10 CFR 50.55a - continued

- Content
  - Must accurately cite specific Code requirement
    - » Edition, Addenda
    - » Section, Subsection, and Paragraph
  - Must accurately cite specific provision of regulations
    - » Alternatives, hardship, or impractical
  - Identify or list applicable components, systems, structures, welds
  - Clear/concise basis for each relief or alternative
  - Describe hardship in detail, fully explain impracticalities
  - Provide drawings where clarity in request is helpful
  - References to earlier submittal for current 10-yr interval

10 CFR 50.59

- Purpose
  - Used to Determine Whether Prior NRC Review and Approval is Necessary Before Licensee Makes:
    - » Changes to facility as described in Safety Analysis Report
    - » Changes in procedures as described in Safety Analysis Report
    - » Test/experiments not described in Safety Analysis Report
  - When Prior Approval Required, Submit Application for Amendment per 10 CFR 50.90

Exemptions: 10 CFR 50.12

- Criteria
  - Must meet one or more special circumstances:
    - » Application of regulation in particular circumstances conflicts with other rules or requirements of NRC
    - » Application of regulation in particular circumstances would not serve the underlying purpose of rule or is not necessary to achieve underlying purpose of rule
    - » Compliance would result in undue hardship or other costs significantly in excess of those incurred by others similarly situated
    - » Exemption would result in benefit to public health and safety that compensates for any decrease in safety that may result from granting the exemption
    - » Exemption would provide only temporary relief from applicable regulation and licensee or applicant has made good faith efforts to comply with regulation
    - » Other material circumstances present not considered when the regulation was adopted for which the exemption would be in public's best interest

Exemptions: 10 CFR 50.12 - continued

- Content
  - Deterministic safety assessment
  - Risk-inform support (optional)
  - Environmental considerations
  - Significant Hazards Determination not required
  - Address how one or more of criteria is met
- Approval
  - Following EA notice in Federal Register
  - NRC Policy: reluctance to change rules by exemptions

Petition for Rulemaking (2.802)

- Content
  - General solution to specify problem or substance or text of proposed regulation or amendment, or specify regulation to amend or revoke
  - Grounds for/interest in action requested
  - Statement of specific issues involved, views or arguments on those issues, relevant data involved, and other pertinent information
  - Specific cases where current rule is unduly burdensome, deficient, or needs to be strengthened
- Timing
  - Submittal deficiency letter from NRC w/in 30 days of receipt
  - Petitioner response to deficiency letter w/in 90 days

Emergency License Amendment: 50.91

- Criteria
  - Must meet all License Amendment criteria from 50.91 and 50.92
  - Failure to act on request would result in
    - » Nuclear power plant shutdown
    - » Prevention of resumption of operation or increase in power up to licensed level
  - Issue without prior notice and opportunity for hearing or public comment ONLY if change would NOT involve significant hazards consideration

Emergency License Amendment: 50.91 - continued

- Content
  - License Amendment content plus
    - » Explanation of why emergency situation occurred
    - » Explanation of why situation could not be avoided
  - Facts must match NOED request information (if NOED issued)
  - NRC publishes notice for opportunity for hearing and public comment after issuance per 2.106
- Timing
  - Amendment not issued if failure to be timely created the emergency
  - Request must be submitted w/in 48 hours if NOED issued

Notice of Enforcement Discretion

- Content (Policy – Inspection Man. 9900, 6/29/99)
  - Tech Spec or License Condition to be violated
  - Description of events leading to request
  - Safety basis: evaluation of significance and potential consequences
  - Basis that noncompliance will not be detriment to public health and safety, does not involve USQ or significant hazard consideration
  - Basis that noncompliance will not involve adverse consequences to environment

Notice of Enforcement Discretion - continued

- Content (con't)
  - Identify compensatory measures, actions taken to avoid noncompliance, actions to avert/alleviate the emergency
  - Justify duration of noncompliance
  - Approval of appropriate review committee

Notice of Enforcement Discretion – continued

- For plant startup: must meet one of 3 criteria
  - Equipment/system does not perform safety function in the mode in which operation is to occur
  - Safety function performed by equipment/system is of only marginal safety benefit, and remaining in current mode increases likelihood of an unnecessary plant transient
  - TS or other license conditions require a test, inspection, or system realignment that is inappropriate for the particular plant conditions, in that it does not provide a safety benefit or may, in fact, be detrimental to safety in the particular plant condition.

Notice of Enforcement Discretion – continued

– Severe weather requests covered by NRC AL 95-05, Revision 2

Notice of Enforcement Discretion - continued

- Region Issues NOED for noncompliance
  - Of short duration (<=14 days) from limits of function specified in LCO
  - With an action statement time limit
  - With a surveillance interval or one-time deviation from surveillance requirement
  - When license amendment is not warranted
- NRR Issues NOED for noncompliance
  - With LCO until LCO can be revised by amendment
  - With action statement time limit until license amendment issued to make temporary or permanent
  - With surveillance interval or change to surveillance by license amendment

Notice of Enforcement Discretion - continued

- Timing
  - Must not abuse requirements of 50.91(a)(5)
  - Oral request must be followed by written request w/in 24 hours
  - NRC Approval letter to be issued w/in 2 working days
  - Region issued NOED not to exceed 14 days
  - Exigent TS amendment request, if appropriate, w/in 48 hours
  - Exigent amendment issued w/in 4 weeks
- References
  - NRC Administrative Letter 95-05, Revision 2
  - NRC Inspection Manual Part 9900, NOEDs, 6/29/99
  - NUREG-1600, NRC Enforcement Policy

General Submittal Concepts/Guidance

- Know the Specific Regulations Affected
- Use Flexibility Allowed by the Regulations
- Keep PM Aware of What is Happening at Plant
- Keep PM Up-to-date With What You Need
- Be Clear in What you are Asking of the Staff
- Submit Requests Early, Allowing Adequate Time for Staff Review

General Submittal Concepts/Guidance - continued

- Provide Future Licensing Needs to Staff Well Before Next Outage
- Plan Ahead for Sholly Notice Period
- Minimize Complexity of the Requests
- Cite Precedents
- Consider Safety Evaluation Perspective
- Provide Complete, Well Written, Thorough, High Quality Submittals
- Provide Copies of Licensing Submittals to PM by Mail and Electronically
- Be Prepared to Interact Promptly with the Staff

BREAKOUT SESSION #2
REVIEW OF ACTUAL SUBMITTAL

Summary of ComEd Application for Technical Specification Amendment to Change Containment Cooling Service Water Requirements for Control Room Emergency Ventilation System (CREVS) support

Discussed at the October 6th and 7th NRC/ComEd Workshop

Proposed revision to Operating License Appendix A, Technical Specifications

The original letter dated May 20, 1999, transmitted a proposed revision to the Operating License Appendix A, Technical Specifications (TS) to identify specific Containment Cooling Service Water (CCSW) equipment requirements to support the Control Room Emergency Ventilation System (CREVS) as required by TS Section 3/4.8.D. The original letter also requested a change to TS Section 3/4.5.C.2 to reflect the minimum suppression chamber water level to ensure proper operation of the low pressure Emergency Core Cooling System (ECCS) pumps. The letter requested approval by October 1, 1999.

The Attachments identified a description of the proposed changes, marked up TS pages, evaluation of No Significant Hazards Consideration and an Environmental Assessment.

On August 10, 1999, the NRC was informed by the Station staff that the requested revision to TS Section 3/4.5.C.2 was necessary to support the Outage Schedule for a refueling outage that was scheduled to begin on October 1, 1999.

During telephone calls on August 30, 1999, and September 2, 1999, the NRC discussed a number of questions related to the submittal. Specifically, the NRC requested information on the CREV and CCSW systems and their interrelationships and power supplies. This information was provided in a letter to the NRC dated September 8, 1999.

On September 14 and 15, 1999, the NRC requested that the wording of requested change to the TS be changed to ensure clarity. The original change included the addition of a footnote to the CCSW Limiting Condition for Operation describing the specific CCSW requirements for CREVS support. This information and new markup of the TS pages was transmitted to the NRC on September 16, 1999.

On September 20, 1999, the NRC informed ComEd that the changes made to the TS pages were incomplete in that the Operational modes specified in the revised TS were incomplete because the TS LCO 3.8.A.2 did not specify all modes for which the specification applied. The revised LCO information was transmitted to the NRC on September 20, 1999.

Additional discussions were held with the NRC on September 25 and 27, 1999, to discuss concerns the NRC had with the reduction in available power supplies. In support of CREVS operability, and in light of the fact that only one CCSW pump would now be required, the NRC considered this to be a increase in risk and requested ComEd to verify

BREAKOUT SESSION #2
REVIEW OF ACTUAL SUBMITTAL

Station Blackout Diesels were available. This request was made to ensure multiple electrical supplies were available to various CCSW pumps. The action to verify SBO diesel availability prior to relying on the reduced number of CCSW pumps was agreed to on September 27, 1999.

The TS change was approved on October 1, 1999.

References:

1) Letter from J.M. Heffley (ComEd) to NRC dated May 20, 1999, "Application for Amendment to Appendix A, Technical Specifications (TS), 3/4.8.D "Containment Cooling Service Water" and Technical Specification 3.5.C "Suppression Chamber"
2) Letter from J.M. Heffley (ComEd) to NRC dated September 8, 1999, "Supplemental Information to the Application for an Amendment to the Technical Specifications"
3) Letter from R.M. Krich (ComEd) to NRC dated September 16, 1999, "Supplement to Application for Amendment to Appendix A, Technical Specifications (TS), 3/4.8.D "Containment Cooling Service Water" and Technical Specification 3.5.C "Suppression Chamber"
4) Letter from J.M. Heffley (ComEd) to NRC dated September 20, 1999, "Additional Supplement to Application for Amendment to Appendix A, Technical Specifications (TS), 3/4.8.D "Containment Cooling Service Water" and Technical Specification 3.5.C "Suppression Chamber"

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ComEd

September 8, 1999

PSLTR# - 99-0066

U.S. Nuclear Regulatory Commission
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Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Supplemental Information to the Application for an Amendment to the Technical Specifications

Reference: Letter from J.M. Heffley (ComEd) to USNRC "Application for Amendment to Appendix A, Technical Specifications (TS), 3/4.8 "Containment Cooling Service Water" and Technical Specification 3.5.C. "Suppression Chamber" dated May 20, 1999

In the reference letter, Commonwealth Edison (ComEd) Company proposed to amend Appendix A, Dresden Nuclear Power Station (Dresden) Technical Specification of Facility Operating Licenses DPR-19 and DPR-25. Specifically, one of the changes ComEd proposed was to clarify the minimum Containment Cooling Service Water (CCSW) equipment required to support operation of the Control Room Emergency Ventilation System (CREVS) as required by Technical Specification Section 3/4.8.D.

During telephone calls conducted on August 30 and September 2, 1999, the NRC asked a number of questions concerning the amendment request. The attachment to this letter provides our response to those questions. Attachment A contains a description of the CREVS system with support system interrelationships, our response to each question the NRC posed, and a description on how the Improved Standard Technical Specifications would address the CREVS/CCSW relationship. Figure A provides a one line diagram of the Main Control Room Ventilation Systems and Figure B provides a one line diagram of the one division of Unit 2 CCSW. ComEd believes that the information provided supports our original conclusion that the definition of a CCSW subsystem as it applies to CREVS operation can be defined as one CCSW pump. This is also supported by our review of NUREG 1433, Revision 1, "Standard Technical Specifications – General Electric Plants, BWR/4." Additionally, as stated in the Reference, this proposed amendment does not create a change to the significant hazards analysis.

As stated in the reference, this amendment request is required by October 1, 1999 in order to support our upcoming refuel outage on Unit 2 (D2R16).

Should you have any questions concerning this letter, please contact Mr. D.F. Ambler at (815) 942-2920, extension 3800.

Respectfully,

Preston Swafford

P. Swafford
Station Manager
Dresden Nuclear Power Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station

Attachments: System Descriptions and Response to Questions

ATTACHMENT

SYSTEM DESCRIPTION AND RESPONSE TO QUESTIONS

Control Room Ventilation System Description

The Control Room HVAC System is comprised of two trains. Figure A provides a one line diagram of the Main Control Room Ventilation Systems and Figure B provides a one line diagram of the one division of Unit 2 CCSW.

Non-safety related Train A normally provides HVAC for the Control Room Emergency Zone. Train A consists of an Air Handling Unit (AHU) and two 50% capacity 45-ton chilled water compressors. Plant Service Water cools these compressors. The electrical power is provided by non-safety related 480v Motor Control Center (MCC) 26-4. In accordance with station procedures. Train A can be manually loaded to Emergency Diesel Generators or the Station Blackout (SBO) Diesel Generators to provide cooling to the Control Room Emergency Zone.

Train B is the standby safety-related portion of the Control Room HVAC System, which is comprised, of one AHU, one 90- ton Refrigeration Condensing Unit (RCU), and the Charcoal Air Filtration Unit (AFU). Train B, the CREVS System, is a single train system with no designed redundancies. This Train was installed to meet the intent of NUREG 0737 Item III.D.3.4, "Control Room Habitability." and accepted by the NRC. The Plant Service Water from both Unit 2 and Unit 3 supplies the normal cooling water. The Containment Cooling Service Water System (CCSW) serves as the standby cooling water source for the RCU, with four pumps normally available to support the containment cooling requirements in Modes 1,2, and 3. Train B is powered by the safety related 480v MCC Bus 29-8 and receives emergency power from the Unit 2 Emergency Diesel Generator (EDG). MCC 29-8 can also receive power from the SBO diesels and the Unit 3 EDG via the Unit 2 and 3 4kv cross-tie.

Both HVAC Trains A and B provide airflow distribution through a common duct system. This design allows either the Train B AHU or Train A AHU (manual power backfeed is required) to support operation of the AFU.

The Control Room Emergency Zone has been reduced by modifications in 1997 that resulted in allowing the removal of the Auxiliary Computer Room from the Control Room Emergency Zone, therefore, removing a large heat load. One CCSW pump can provide sufficient cooling to maintain the Control Room within design temperature requirements of 70 to 80 degrees F. This was confirmed during post-modification testing after removal of the Auxiliary Computer Room from the Control Room Emergency Zone. The test also demonstrated that, with both Control Room HVAC trains shutdown (e.g. no cooling provided), the Control Room temperatures only rose to 80 degrees F after 4 hours.

ATTACHMENT

SYSTEM DESCRIPTION AND RESPONSE TO QUESTIONS

By design, the Control Room HVAC System will have sufficient cooling water even if Units 2 and 3 experience a Loss of Offsite Power (LOOP), one unit (Unit 2 or 3) experiences a Design Basis Accident (DBA) and Unit 2 was in a Refueling Outage. Train B is designed to receive cooling water and power by Emergency Diesels or the SBO Diesel while experiencing this type of scenario. The probability of the type of scenario occurring is extremely low.

NRC Questions and Responses

Question 1

Attachment A, Safety Analysis of the Proposed Change," first paragraph states that, in Modes 1, 2, and 3, two trains of CCSW are required to be operable for containment cooling, and will therefore continue to be operable to support the CREVS. Wouldn't this only be true if Unit 2 is In Modes 1, 2, or 3 and not cover the conditions where Unit 2 is in an outage? The No Significant Hazards states that the proposed change "does not reduce the availability of systems required to mitigate accident conditions..." despite this, there appears to be a significant reduction in the availability of redundant support systems for Train B CREVS when Unit 2 is in an outage.

Response:

The statement in the 1st paragraph of the license amendment request is only applicable while in Modes 1, 2, and 3. Only one CCSW pump is required to provide backup cooling water support for the CREVS. The purpose of the statement was to reinforce that more than the required number of CCSW pumps for CREVS support are required in those Modes.

With respect to the second question, there is not a significant reduction in the availability of support systems, CCSW is still available and therefore the number of support systems remains unchanged. CREVS is a single train system that was never designed with redundant safety-related support systems. For example, its primary cooling water supply is provided via the plant service water system. This primary water supply can be from either Unit 2 or Unit 3. Unless offsite power is lost to both Unit 2 and Unit 3, plant service water will be available to supply the CREVS RCU. Also only one CCSW pump is required to support this system. Therefore, as stated within the No Significant Hazards Consideration, there is no significant reduction in the <u>required</u> equipment necessary to mitigate the consequences of an accident.

ATTACHMENT

SYSTEM DESCRIPTION AND RESPONSE TO QUESTIONS

Question 2

Please describe the HVAC support system redundancy (including onsite power) that will be maintained when Unit 3 is in Modes 1, 2 or 3 and Unit 2 is in an outage.

Response:

The normal HVAC system for the control room AC (Train A) is provided by two 50% capacity 45-ton chilled water compressors which are non-safety related and not normally aligned to receive power from an EDG. Train B, the CREVS system, is a single train system with no designed redundancies (i.e. there is only a single RCU unit and a single filtration unit). Support systems include non-safety related cooling water to the RCU, which is normally provided from either Unit 2 or Unit 3 plant service water systems. CCSW from Unit 2 provides a safety–related cooling water supply. Electrical power is provided via a safety-related bus, normally fed by the Unit 2 EDG upon loss of power.

Question 3

Describe any additional operator actions or system realignments that are required to activate Train B CREVS with the reduced number of available support systems described above. How much time is available to the operator to align Train B CREVS after an accident.

Response:

No additional operator actions are required or system realignments necessary. Only one CCSW pump is necessary to supply cooling water if neither Unit 2 or Unit 3's service water system is available. The CCSW supply to CREVS is located just outside of the control room. Operations is required, in accordance with approved station procedures, to manipulate the CCSW supply to CREVS within 40 minutes.

Question 4

What is the importance of CREVS in the IPE (e.g., what is the impact of an extended loss CR cooling) What assumptions were made in the IPE with regard to the availability of the CCSW pumps to support CREVS, and how would the proposed reduction In redundancy affect these assumptions? How would this change CDF and LERF?

Response:

The CREVS is not modeled in the IPE as it is not a core damage mitigation system. However, the probability of a dual unit LOOP and LOCA while Unit 2 is in a refueling outage coupled with a CCSW pump failure to start is so low that it can be concluded that the lack of a second CCSW pump would have insignificant impact on either CDF or LERF.

Question 5:

Attachment A, "Bases for the proposed change," first paragraph, states the TS should specify "operable pump" instead of "operable subsystem" because flow from CCSW to CREVS does not flow through the LPCI/CCSW heat exchanger. Describe the CCSW lineup/operation if the rest of the CCSW subsystem is inoperable. Specifically, if there is no flow through the LPCI/CCSW heat exchanger, can the CCSW pump (which is rated for 3500 gpm) operate long term providing just 121 gpm to CREVS? Should the TS specify "operable flow path?"

Response:

A minimum flow path of 350 gpm is established and maintained when the flowpath through the LPCI/CCSW heat exchangers is not available. A CCSW pump would run on the minimum flow path until such time that offsite power would be restored and service water could be re-established. It should be noted that the CCSW system would only be needed to support the CREVS RCU if a LOOP occurred on BOTH units with a LOCA on Unit 2. ComEd believes that the words "...and an operable flow path..." are redundant to the TS definition of OPERABLE/OPERABILITY in that no pump can perform its intended function without an operable flow path.

Question 6:

Describe the CREVS response to a fuel handling accident. How long is CREVS Train B required to operate after FHA?

Response:

The CREVS system filtration unit will perform its intended function independent of whether the RCU or RCU support systems (such as CCSW) are operable. The RCU is required to maintain design temperatures in the control room and, as such, is not considered in the mitigation of the FHA.

ATTACHMENT

SYSTEM DESCRIPTION AND RESPONSE TO QUESTIONS

Improved Standard Technical Specifications (ITS)

ITS perspective of Control Room AC System Operability with regard to the CCSW System can be found through reference to Section 3.7.1, "Containment Cooling Service Water," and the Basis for Section 3.7.5, "Control Room Emergency Ventilation Air Conditioning System."

In ITS, no Limiting Conditions of Operation or Action Statements with regards to the proposed CCSW Operability during MODE * would be required. In that MODE, CCSW is a support system for system(s) that have separate Technical Specification(s). Therefore, in order for the supported systems to meet the definition of OPERABILITY, its supporting system and/or components would have to be OPERABLE. The specific requirement for the support function of CCSW to the Control Room AC system will be placed in a Technical Requirement Manual (TRM).

The specific requirement for CCSW regarding the OPERABILITY of the Control Room AC system is the ability of CCSW to supply the appropriate amount of cooling water flow to the system. This is accomplished by one CCSW pump. Therefore only one CCSW pump would be identified in the TRM.

Conclusion:

Train B, the CREVS System, is a single train system, installed to meet the intent of TMI Action Item III.D.3.4. This train has no design redundancies (e.g. there is only a single RCU unit and a single filtration unit). It was never the intent of ComEd, as required by the NRC, to provide multiple pumps to perform a backup cooling water function for the CREVS System. As such, this level of redundancy is not required to support this single train system.

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701

ComEd

September 16, 1999

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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating Licenses Nos. DPR-19 and DPR-25
<u>NRC Docket Nos. 50-237 and 50-249</u>

Subject: Supplement to Application for Amendment to Appendix A, Technical Specifications (TS), 3/4.8 "Containment Cooling Service Water" and Technical Specification 3.5.C. "Suppression Chamber"

References: Letter from J. M. Heffley (ComEd) to USNRC, "Application for Amendment to Appendix A, Technical Specifications (TS), 3/4.8 - Containment Cooling Service Water and Technical Specification 3.5.C. - Suppression Chamber" dated May 20, 1999

In the referenced letter, in accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company proposed, in part, to amend Appendix A, Dresden Nuclear Power Station (Dresden) Technical Specification Section 3.8.A of Facility Operating Licenses DPR-19 and DPR-25. The purpose of that license amendment request was to identify the specific Containment Cooling Service Water (CCSW) equipment required to support operation of the Control Room Emergency Ventilation System (CREVS) as required by Technical Specification Section 3/4.8.D.

During telephone conversations held on September 14, and September 15, 1999, the NRC requested the wording ComEd proposed to define CCSW equipment requirements in support of CREVS be restructured to ensure clarity. ComEd agrees with the NRC's suggestions and the attachment to this letter provides the revised wording for Appendix A, Section 3.8.A of the Operating License and associated Bases. The Updated Final Safety Analysis Report will be revised to clearly reflect the flowpath and minimum flow requirements of CCSW in support of the CREVS.

ComEd has reviewed the original No Significant Hazards Consideration for this license amendment request and has determined that this clarification does not change its conclusion.

This proposed supplement to the license amendment request has been reviewed and approved in accordance with ComEd procedures.

ComEd requests NRC approval of this license amendment request by October 1, 1999.

ComEd is notifying the State of Illinois of this supplement to its original application for amendment by transmitting a copy of this letter and its attachment to the designated state official.

Please direct any questions you may have concerning this submittal to Mr. Dale Ambler, Regulatory Assurance Manager (815) 942-2920 extension 3800.

Respectfully,

R.M. Krich
Vice President – Regulatory Services

Attachment:

A. Marked-Up Technical Specification Pages

cc: Regional Administrator – NRC Region III
Senior Resident Inspector – Dresden Nuclear Power Station

bcc: NGG Senior Vice President – ComEd
NGG Senior Vice President – Nuclear Operations – ComEd
Vice President Regulatory Services
Site Vice President – Dresden Nuclear Power Station
Station Manager – Dresden Nuclear Power Station
Decommissioning Plant Manager (U1 Only)
Regulatory Assurance Manager – Dresden Nuclear Power Station
Regulatory Assurance Manager – Quad Cities Nuclear Power Station
Site Engineering Manager – Dresden Nuclear Power Station
Operations Manager – Dresden Nuclear Power Station
Training Manager – Dresden Nuclear Power Station
Project Manager, NRR (Unit 2/3) – Dresden Nuclear Power Station
Director Dresden/Quad Cities Licensing and Compliance – ComEd
Office of Nuclear Facility Safety – IDNS
Winston and Strawn
DCD, Licensing (Hard Copy)
DCD, Licensing (Electronic Copy)
Dresden Regulatory Assurance, Subject File
Dresden Nuclear Licensing Administrator - ComEd
K. Beverly, Licensing Engineer – Dresden Nuclear Power Station
NSRB Coordinator – Dresden Nuclear Power Station

ATTACHMENT

MARKED UP CHANGES TO THE TECHNICAL SPECIFICATIONS

PAGES

3/4.8-1

B3/4.8-1

Commonwealth Edison Company
Dresden Generating Station
6500 North Dresden Road
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Tel 815-942-2920

ComEd

September 20, 1999

JMHLTR: No. 99-0107

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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating Licenses Nos. DPR-19 and DPR-25
<u>NRC Docket Nos. 50-237 and 50-249</u>

Subject: Additional Supplement to Application for Amendment to Appendix A, Technical Specifications (TS), 3/4.8 "Containment Cooling Service Water" and Technical Specification 3.5.C "Suppression Chamber"

References: (a) Letter from J. M. Heffley (ComEd) to USNRC, "Application for Amendment to Appendix A, Technical Specifications (TS), 3/4.8 - Containment Cooling Service Water and Technical Specification 3.5.C. -Suppression Chamber, dated May 20, 1999

(b) Letter from R.M. Krich to USNRC, "Supplement to Application for Amendment to Appendix A, Technical Specifications (TS), 3/4.8 – Containment Cooling Service Water and Technical Specification 3.5.C - Suppression Chamber, dated September 20, 1999

In the referenced letters, in accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company proposed, in part, to amend Appendix A, Dresden Nuclear Power Station (Dresden) Technical Specification Section 3.8.A of Facility Operating Licenses DPR-19 and DPR-25. The purpose of that license amendment request was to identify the specific Containment Cooling Service Water (CCSW) equipment required to support operation of the Control Room Emergency Ventilation System (CREVS) as required by Technical Specification Section 3/4.8.D.

During telephone conversations held on September 20,1999, the NRC requested additional clarification to the wording ComEd proposed to define CCSW equipment requirements in support of CREVS. Enclosed in the attachment to this letter is the revised wording for Appendix A, Section 3.8.A of the Operating License.

ComEd has reviewed the original No Significant Hazards Consideration for this license amendment request and has determined that this clarification does not change its conclusion.

This proposed supplement to the license amendment request has been reviewed and approved in accordance with ComEd procedures.

ComEd requests NRC approval of this license amendment request by October 1, 1999.

ComEd is notifying the State of Illinois of this supplement to its original application for amendment by transmitting a copy of this letter and its attachment to the designated state official.

Please direct any questions you may have concerning this submittal to Mr. Dale Ambler, Regulatory Assurance Manager (815) 942-2920 extension 3800.

Respectfully,

J.M. Heffley
Site Vice President
Dresden Station

Attachment:

A. Marked-Up Technical Specification Page

cc: Regional Administrator – NRC Region III
Senior Resident Inspector – Dresden Nuclear Power Station

ATTACHMENT

MARKED UP CHANGES TO THE TECHNICAL SPECIFICATIONS

PAGE

3/4.8-1

3/4.8-3

Commonwealth Edison Company
Dresden Generating Station
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Tel 815 942-2920

Subject

ComEd

May 20, 1999

JMHLTR: #99-0062

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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating Licenses Nos. DPR-19 and DPR-25
<u>NRC Docket Nos. 50-237 and 50-249</u>

Subject: Application for Amendment to Appendix A, Technical Specifications (TS), 3/4.8 "Containment Cooling Service Water" and Technical Specification 3.5.C. "Suppression Chamber"

References: A) Letter from J.P.Dimmette (ComEd) to USNRC, "Request for License Amendment Change to Various Acceptance Values to Reconcile with Design Values "dated May 18, 1998

Pursuant to 10 CFR 50.90, Commonwealth Edison (ComEd) Company proposes to amend Appendix A, Dresden Nuclear Power Station (Dresden) Technical Specification Section 3.8.A of Facility Operating Licenses DPR-19 and DPR-25. The purpose of this amendment request is to identify specific Containment Cooling Service Water equipment required to support operation of the Control Room Emergency Ventilation System (CREVS) as required by Technical Specification Section. 3/4.8.D.

Additionally, ComEd proposes to revise the Technical Specifications 3.5.C.2 and Surveillance Requirement 4.5.C.2 to reflect the required minimum suppression chamber water level to ensure proper operation of the low pressure Emergency Core Cooling System (ECCS) pumps. The proposed Technical Specification Amendment is subdivided as follows:

1. Attachment A gives a description and safety analysis of the proposed changes.

2. Attachment B includes the proposed changes to the Technical Specification pages, including marked-up versions of the current pages.

3. Attachment C describes ComEd's evaluation performed in accordance with 10 CFR 50.92(c), which confirms that no significant hazards consideration is involved. In addition, ComEd's Environmental Assessment Applicability Review is included.

4. Attachment D provides the Environmental Assessment.

This proposed Technical Specification amendment has been reviewed and approved by ComEd On-Site and Off-Site Review in accordance with ComEd procedures.

why? ComEd requests NRC approval of this request by October 1, 1999

ComEd is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated state official.

Please direct any questions you may have concerning this submittal to Dale Ambler, Regulatory Assurance Manager (815) 942-2920 extension 3800.

Respectfully,

J. M. Heffley
Site Vice President
Dresden Nuclear Power Station

Attachments:

A. Description and Safety Analysis of the Proposed Changes
B. Marked-Up Technical Specification Pages
C. Evaluation of Significant Hazards Considerations and Environmental Assessment Applicability Review
D. Environmental Assessment

cc: Regional Administrator – NRC Region III
Senior Resident Inspector – Dresden Nuclear Power Station

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

[TITLE]

Description of the Current Requirement

Dresden Nuclear Power Station Technical Specification 3.8.A specifies the applicability, limiting conditions for operation and action statements for the Containment Cooling Service Water System (CCSW). The specific requirement states that:

> At least the following independent containment cooling service water (CCSW) subsystems, with each comprised of:
>
> 1. Two OPERABLE CCSW pumps, and
> 2. an OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water:
>   a. Through one LPCI heat exchanger, and separately,
>   b. To the associated safety related equipment,
>   shall be OPERABLE.
>
> 1. In OPERATIONAL MODE(s) 1, 2 and 3, two subsystems.
> 2. In OPERATIONAL MODE *, the subsystems (s) associated with subsystems loops and components required OPERABLE by Specification 3.8.D.

Bases for the Current Requirement

The CCSW systems are designed to remove heat from the containment, reduce containment pressure and restore suppression pool temperature following a postulated Loss-of-Coolant Accident (LOCA). This is accomplished by having two separate, two pump, flow (subsystems) loops. Each pair of CCSW pumps (two per loop) draws water from the cribhouse suction bays (ultimate heat sink) via separate supply piping. Two CCSW pumps discharge into a common header which routes the cooling water to that loop's associated heat exchanger. At the heat exchanger, heat is transferred from the low pressure coolant injection (LPCI) subsystem to the CCSW system, and subsequently, to the ultimate heat sink.

CCSW also provides cooling water to the Refrigeration Control Unit (RCU) of the Main Control Room Emergency Ventilation System (CREVS). Technical Specification 3.8.D provides the limiting conditions for operation (LCOs), applicability and action statements for the CREVS. The CREVS assures that, during the accident conditions, the main control room remains habitable for the operators as well as assures the required heat is removed from the control room atmosphere in accordance with Surveillance Requirement 4.8.D.1. CREVS is a single train filtration system that can be powered from the Unit 2 Emergency Diesel Generator. Normal cooling water to the CREVS RCU is provided by plant service water from either Unit 2 or Unit 3. Plant service water is non-safety related and would not normally be powered by the Emergency Diesel Generator post-accident. Unit 2 CCSW provides a 121 gpm water supply to the CREVS RCU to assure proper cooling of the RCU compressor. CCSW pumps have a design capacity of 3500 gpm per pump. Therefore, only a small fraction of flow from one CCSW pump is required to assure proper performance of the RCU.

<u>Description of the Proposed Change</u>

In accordance with 10CFR50.90, ComEd proposes to clarify the APPLICABILITY of Specification 3.8.A to note that only one CCSW pump is required to support RCU operation for the CREVS in OPERATIONAL MODES 1, 2, 3 and *. This change is accomplished by adding new footnote (a) to APPLICABILITY statement which states: "*Any one of four Unit 2 CCSW pumps is required to support CREVS RCU operation.*" Another change will be the removal of OPERATIONAL MODE * as a separate line item and inclusion of OPERATIONAL MODE * with modes 1, 2, and 3. The second reference to "two subsystems" in the APPLICABILITY statement has been deleted thereby requiring replacement of "At least the following" with the word "Two" in the opening sentence of the LCO. Therefore, the revised specification is proposed to read:

> Two[a] independent containment cooling service water (CCSW) subsystems, with each subsystem comprised of:
>
> 1. Two OPERABLE CCSW pumps, and
> 2. An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water:
>   a. Through one LPCI heat exchanger, and separately,
>   b. To the associated safety related equipment,
>   shall be OPERABLE.
>
> In OPERATIONAL MODE(s) 1, 2, 3, and *.

Footnote (a) will be placed below the * footnote and will read

a Any one of four Unit 2 CCSW pumps is required to support CREVS RCU operation.

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

Revision of the TS 3.8.A Action 2 is required to reflect that a subsystem as described in the APPLICABILITY is overly restrictive and that, when in OPERATIONAL MODE *, one CCSW pump must be operable to support the CREVS RCU.

Current TS 3.8.A Action 2 states:

> "In OPERATIONAL MODE * with the CCSW subsystem which is associated with the safety related equipment required OPERABLE by Specification 3.8.D inoperable, declare the associated safety related equipment inoperable and take the ACTION required by Specification 3.8.D."

ComEd proposes to replace the "CCSW subsystem" in ACTION 2 with "CCSW pump" to be consistent with proposed footnote (a). The revised action will read:

> "In OPERATIONAL MODE * with the CCSW pump which is associated with the safety related equipment required OPERABLE by Specification 3.8.D inoperable, declare the associated safety related equipment inoperable and take the ACTION required by Specification 3.8.D."

Bases for the proposed change

Irrespective of OPERATIONAL MODE, the CREVS RCU compressor requires 121 gpm from a cooling medium, either plant service water or Unit 2 CCSW. The capacity of each CCSW pump is 3500 gpm. Therefore, only one CCSW pump is required operable to support CREVS operation. If plant service water is unavailable, one CCSW pump can supply the required cooling water flow to the RCU. CCSW is a load which is expected to be connected to the Emergency Diesel Generator post-accident. Flow from CCSW to the CREVS RCU does not pass through the CCSW heat exchanger. Therefore, requiring an OPERABLE "subsystem" as defined in Specification 3.8.A is overly restrictive and not consistent with design requirements. Unit 3 CCSW does not provide any water to the CREVS RCU, therefore the revised footnote reflects that design.

Need for the proposed change

Current Specification 3.8.A does not clearly state that only one CCSW pump is required to support CREVS operation. The LCO 3.8.A requires two CCSW pumps and an operable flow path capable of taking suction from the ultimate heat sink and transferring water through one heat exchanger or separately to the associated safety related equipment. The CREVS RCU compressor system is one system that uses CCSW water as the cooling medium. Additional provisions of the LCO require 1) Two subsystems operable in OPERATIONAL MODE(s) 1,2 and 3 and, in OPERATIONAL MODE *, the subsystems associated with the subsystem/loops and components required OPERABLE by Specification 3.8.D (CREVS).

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

As previously stated, the CREVS is a single train system equipped with one RCU. The RCU needs 121 gpm to cool its compressor. Therefore, one CCSW pump is required to fulfill this function irrespective of the OPERATIONAL MODE. The current specification requires 2 subsystems operable in OPERATIONAL MODE 1, 2 and 3 for containment cooling purposes and should not impose overly restrictive requirements for CCSW support of the CREVS system.

The second part of the APPLICABILITY statement of current LCO 3.8.A provides some clarity that there are different requirements of the CCSW system to support the CREVS RCU versus the containment cooling function. However, the requirements to have one CCSW pump operable for CREVS is not mode dependent as suggested by the current technical specification. The proposed footnote (a) states the requirements clearly.

Additionally, Technical Specification 3.8.A Action 2 has been appropriately revised to reflect that a subsystem as described in the current APPLICABILITY is not required in OPERATIONAL MODE *. The containment cooling function is not required when the reactor is only in OPERATIONAL MODE *. Therefore, one CCSW pump is required operable to support CREVS and Action 2 has been modified to reflect that requirement.

Safety Analysis of the Proposed Change

The CREVS system provides a radiologically controlled environment from which the plant can be operated after a design basis accident. The RCU maintains the temperature in the control room at an acceptable level for the control room operators. Water supplied to the RCU compressor for cooling via service water may not be available post-accident, therefore, Unit 2 CCSW provides a water supply that is available post-accident since CCSW pumps are expected loads on the Emergency Diesel Generators post-accident. The proposed change has a minimal effect on safety since, in OPERATIONAL MODE(s) 1, 2 and 3, two subsystems will continue to be required to support the containment cooling function of CCSW. Therefore, at least more than one CCSW pump will continue to be operable to provide support of the CREVS. In OPERATONAL MODE *, the proposed change provides clarity that only one CCSW pump is required operable to support the CREVS. The proposed changes are consistent with the current design and remove the appearance of overly restrictive provisions.

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

Suppression Chamber level (from ≥ 8 feet to ≥ 10 feet, 4 inches.)

Description of the Current Requirement

Pursuant to the provisions of the 10CFR50.90, ComEd proposes to revise TS Sections 3.5.C.2 and 4.5.C.2, "Suppression Chamber." This amendment proposes to raise the allowable level in the suppression chamber while operating in OPERATIONAL MODEs 4 or 5 from ≥8 feet to ≥10 feet, 4 inches (10' 4").

Current TS Section 3.5.C.2 requires the suppression chamber to be OPERABLE in OPERATIONAL MODE(s) 4 and 5 with a contained volume equivalent to a water level of ≥ 8 feet above the bottom of the suppression chamber. An exception is provided for OPERATIONAL MODE 5, allowing removal of all water from the suppression pool when the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specification 3.10.G and 3.10.H (reactor vessel and spent fuel pool water level requirements during refuel operations).

The current water level requirement in OPERATIONAL MODE(s) 4 and 5 is based on providing adequate NPSH for a single ECCS pump start. However, under certain scenarios, there could be an autostart of as many as 6 low-pressure ECCS pumps operating at unthrottled flows. Thus, the current technical specification allowable level is non-conservative, and may not provide the required ECCS NPSH during an event. Therefore, ComEd proposes to raise the suppression pool level requirements from ≥8 feet to ≥ 10 feet, 4 inches.

Bases for the Current Requirement

The current requirement for suppression chamber level while in OPERATIONAL MODEs 4 and 5 is based on ensuring that a sufficient supply of water is available to the Core Spray and Low Pressure Coolant Injection (LPCI) systems in the event of a ~~Loss~~ of Coolant Accident (LOCA). Since pressure suppression is not required below are reactor moderator temperature of 212°F, the minimum suppression pool water volume is based on net positive suction head (NPSH), recirculation volume and vortex prevention. The calculation, which supports this requirement, assumed one low pressure ECCS pump operating at design flow.

Description of the Proposed Change

This change will amend TS Sections 3.5.C.2 and 4.5.C.2 to raise the allowable level in the suppression chamber while in OPERATIONAL MODEs 4 or 5 from ≥8 feet to ≥10 feet, 4 inches. By raising the minimum water level to 10 feet, 4 inches, air entrainment due to vortexing will be prevented should all six low-pressure ECCS pumps auto-start. The low-pressure ECCS pumps are expected to cavitate until the operators act to return the unthottled flow to design flow for each pump. Once at design flow, NPSH is maintained with a suppression pool water level at or above 10 feet, 4 inches.

Safety Analysis of the Proposed Change

The current requirement for suppression pool level during OPERATIONAL MODEs 4 and 5 is ≥8 feet. This requirement assumes one low pressure ECCS pump running at design flow. The potential exists for an auto start signal in OPERATIONAL MODEs 4 and 5 starting as many as 6 low pressure ECCS pumps operating at unthrottled flows, even though one pump would be sufficient to ensure core coverage. The current allowable level is non-conservative, and may not provide the required NPSH or avoid air entrainment into the ECCS pumps during auto-start of 6 ECCS pumps. Initiation of six low pressure ECCS pumps with zero reactor pressure will result in pump cavitation-until the pumps are either shutdown or throttled to design flows. Existing procedures require pump shutdown or throttling of pump flow on indication of cavitation. Dresden ECCS pumps have been tested for short-term cavitation without causing damage which would prevent long term operation. The proposed technical specification change restores margin to ensure the ECCS pumps are protected and are available to perform their design basis function.

Impact on Previous Submittals

ComEd has reviewed the proposed Operating License Amendment request regarding impact on any previous submittals, and has determined that there is no impact on any outstanding previous submittals.

Schedular Requirements

Approval of this TS change is requested by October 1, 1999. These issues either clarify existing requirements or correct TS limits that are non-conservative with respect to the design basis of the plant. Timely approval of this TS change will ensure that the TS reflect the current station design requirements and allows ComEd to close an open operability determination. A 60-day implementation period will provide sufficient time to reflect the changes to the TS in plant procedures, processes and training.

ATTACHMENT B

EVALUATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS

Pages 3/4.5-7

B3/4.5-4

3/4.8-1

3/4.8-3

ATTACHMENT C

EVALUATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10CFR50.92(c). A proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed changes to Technical Specification 3.8.A Limiting Conditions for Operation to clarify that only 1 CCSW pump is required to support operation of the CREVS

The second proposed change is to raise the allowable suppression pool level during OPERATIONAL MODES 4 and 5 to restore margin required to prevent vortexing in the ECCS pump suction.

ComEd has evaluated the proposed Technical Specification Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of Dresden Units 2 & 3 in accordance with the proposed amendment will not:

1) **Involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:**

   The proposed changes to the technical specifications provide clarity in the support system relationship and requirements for the CCSW system support of the CREVS operation. The CCSW system nor the CREVS system are assumed to be accident precursors for previously evaluated accident. Therefore, the proposed changes have no effect on the probability or consequences of accidents previously evaluated.

   The proposed change to the allowable suppression chamber level does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change revises a Technical Specification acceptance value to more conservative value and serves to ensure operability of equipment important to safety. By ensuring equipment availability, the probability or consequences of an accident previously evaluated are not increased. In addition, the proposed changes have no impact on any initial condition assumptions for accident scenarios. Onsite or offsite dose consequences resulting from an event previously evaluated are not affected by this proposed amendment request.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT

ENVIRONMENTAL ASSESSMENT

ComEd has evaluated this proposed operating license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that this proposed license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) the amendment involves no significant hazards consideration.

As demonstrated in Attachment C, this proposed amendment does not involve any significant hazards consideration.

(ii) there is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

As documented in Attachment C, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

ATTACHMENT C

EVALUATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

2) **Create the possibility of a new or different kind of accident from any accident previously evaluated because:**

The proposed changes do not create the possibility of a new or different kind of accident from that previously evaluated. The changes to the CCSW specifications more appropriate reflect the design requirements and clarify the support role of the CCSW system as it relates the CREVS. Neither the CCSW system nor the CREVS will be operated differently with the proposed change. Therefore new or different failure modes will not be created. Therefore, the possibility of new and different accidents has not been created with the proposed change

The proposed change to the suppression pool allowable level restores margin to the Technical Specifications and ensures equipment operability. The proposed change is conservative with respect to current requirements. The proposed amendment does not involve any plant physical changes that would create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) **Involve a significant reduction in the margin of safety because:**

The proposed change to the CCSW technical specification will not result in a significant reduction in the margin of safety. The proposed change has greater consistency with the current design requirements for CSSW support of CREVS operation. Therefore, the margin of safety has been not been altered.

The proposed changes for suppression pool level does not involve a significant reduction in a margin of safety. In fact, the proposed changes restore margin and ensure equipment operability. Since the changes maintain the necessary level of system reliability, they do not involve a significant reduction in the margin of safety.

The proposed amendment for Dresden will not reduce the availability of systems required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations.

ATTACHMENT C

EVALUATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

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TOLEDO EDISON

RICHARD P CROUSE
Vice President
Nuclear
(419) 249 5221

Docket No. 50-346
License No. NPF-3
Serial No. 1159
June 17, 1985

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz
Operating Reactor Branch No. 4
Division of Licensing
United States Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Stolz:

Under separate cover, we are transmitting three (3) original and forty (40) conformed copies of an application for Amendment to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station Unit No. 1.

This application requests that the Davis-Besse Nuclear Power Station Unit 1 Technical Specifications, Appendix A, be revised to reflect the changes attached. The proposed change is in Section 3.7.1.1.

The attachment identifies the proposed changes and its safety evaluation and a significant hazard consideration. The proposed change concerns the Limiting Condition for Operation Action Statement which now requires that with one or more Code Safety Valves inoperable, either restore the valve(s) to OPERABLE status, or reduce power per Table 3.7-1, or shutdown the plant and proceed to COLD SHUTDOWN. This amendment requests to change COLD SHUTDOWN (MODE 5) to HOT SHUTDOWN (MODE 4). The Code Safety Valves APPLICABILITY of this section is MODES 1, 2 and 3; therefore, the valves are not required in HOT SHUTDOWN (MODE 4) and the plant should not be required to enter COLD SHUTDOWN (MODE 5).

Toledo Edison requests that this amendment request be approved and issued by February 28, 1986.

Enclosed is a check for $150 as required by 10CFR170.12(C) for license application.

Very truly yours,

RPC:GAB:RLW

Attachment

cc: DB-1 NRC Resident Inspector
State of Ohio

J U U U 0 3 0 4 7 6 3

APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NO. NPF-3

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NO. 1

Enclosed are forty-three (43) copies of the requested changes to the Davis-Besse Nuclear Power Station Unit No. 1 Facility Operating License No. NPF-3, together with the Safety Evaluation for the requested change.

The proposed changes include Section 3.7.1.1.

By /s/ R. P. Crouse
Vice President, Nuclear

Sworn and subscribed before me this 17th day of June, 1985.

/s/ Laurie A. Hinkle, nee (Brudzinski)
Notary Public -- State of Ohio
My Commission Expires May 16, 1986.

S E A L

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Docket No. 50-346
License No. NPF-3
Serial No. 1159
June 17, 1985

Attachment

I. Changes to Davis-Besse Nuclear Power Station Unit 1, Appendix A Technical Specifications Section 3.7.1.1

A. Time required to Implement. This change is to be effective upon NRC approval.

B. Reason for Change (Facility Change Request 85-0051, Rev. A). With one or more of the Code Safety Valves inoperable and if the plant is shutdown per the Action Statement per Section 3.7.1.1 it must be in Mode 5, COLD SHUTDOWN. The Applicability of the Action Statement is only MODES 1, 2 and 3, therefore, entry into Mode 5 (Cold Shutdown) should not be required.

C. Safety Evaluation
(See Attached)

D. Significant Hazard Consideration
(See Attached)

SAFETY EVALUATION

This FCR is to revise Tech. Spec. 3.7.1.1 action statement (change the words from "COLD SHUTDOWN within the following 30 hours" to "HOT SHUTDOWN within the following 12 hours"). This action statement concerns the Limiting Condition for Operation for the main steam line code safety valves.

The safety function of Tech. Spec. 3.7.1.1 is to ensure overpressure protection for the plant secondary side. The code safety valves are needed to relieve excess steam in the event of various transients such as, loss of load, loss of offsite power, loss of condenser vacuum, etc. Pressure relief is required at the system design pressure of 1050 psig, and the first safety valve bank is set to relieve at this pressure. Additional safety valve banks are set at pressures up to 1100 psig, as allowed by the ASME Code.

Existing Tech. Spec. 3.7.1.1 action statement states that with one or more main steam line code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided that, within 4 hours, either the inoperable valve is restored to OPERABLE status or the High Flux Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. In reviewing the Limiting Condition for Operation for the main steam line safety valves, it was noted that a discrepancy existed. The existing action statement for inoperable safety valves ultimately places the unit in COLD SHUTDOWN (MODE 5). However, the valves are only required to be operable in MODES 1, 2 & 3. Therefore, the action statement should require the plant to go to HOT SHUTDOWN (MODE 4).

During cooldown in MODES 1, 2, & 3 the steam generators reduce the reactor coolant system temperature from operating temperature to ≤ 280°F and the code safety valves must be operable to relieve potential excessive steam generation in the event of an abnormal transient. The Decay Heat Removal System (DHR) is then placed in operation when entering MODE 4 to reduce the reactor coolant temperature to the desired level. The DHR system is required to be operable in MODE 4, this includes safety relief valve DH 4849 (see Tech. Spec. 3.4.2). There is no accident initiating from MODE 4 that would require the operation of main steam code safety valves, therefore, the safety function of Tech. Spec. 3.7.1.1 or the plant is not being degraded by this change.

The 12 hour time requirement to reach the HOT SHUTDOWN condition is consistent with other Tech. Spec. (see T.S. 3.7.1.5, 3.5.2, 3.4.3) that require the plant to go to MODE 4.

Pursuant to the above analysis, it is concluded that the change as proposed does not degrade the safety function of this Tech. Spec. or the plant and therefore, there is no unreviewed safety questions involved.

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SIGNIFICANT HAZARD CONSIDERATION

The attached amendment request to revise the required Shutdown Mode per Action Statement contained within Section 3.7.1.1 from COLD SHUTDOWN to HOT SHUTDOWN does not represent a Significant Hazard Consideration.

Section 3.7.1.1 of the Davis-Besse Technical Specification requires one of the following if one or more of the code safety valves are inoperable:

a. Restore valve to OPERABLE status within 4 hours,
b. Reduce High Flux Trip Setpoint per Table 3.7 (within 4 hours), or
c. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The APPLICABILITY of this Section is for MODES 1, 2 and 3, but the Action Statement requires the plant to be placed in MODE 5 (COLD SHUTDOWN). In MODE 4 (HOT SHUTDOWN) the Code Safety valves are not required to be operable.

In MODE 4 Reactor Coolant System (RCS) temperature is between 280 and 200°F and the Decay Heat Removal (DHR) System is required OPERABLE in MODES 4, 5 and 6. The DHR system provides the decay heat removal and over pressure protection for the RCS. There is no accident in MODE 4 which requires the operation of the Code Safety Valves for mitigation of an accident.

The granting of the request would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated (10CFR50.92(C)(1)).

   The changing of the Action Statement to require the plant to enter HOT STANDBY (MODE 4) and not COLD SHUTDOWN (MODE 5) will not increase the probability or consequences of an accident previously evaluated. The code safeties are not required to be OPERABLE in HOT STANDBY (MODE 4) and would not change any accident previously evaluated.

2. Create the possibility of a new or different kind of accident previously evaluated (10CFR50.92(C)(2)).

   All accidents are still bounded by previous analysis and no new accidents are involved.

3. Involve a significant reduction in a margin of safety (10CFR50.92(C)(3)).

   This amendment request will not reduce the margin of safety assumed in the accident analysis at Davis-Besse.

On the basis of the above, Toledo Edison has determined that the amendment request does not involve a significant hazard consideration.

mj c/2

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TOLEDO EDISON

JOE WILLIAMS, JR.
Senior Vice President—Nuclear
(419) 249-2300
(419) 249-5223

Docket No. 50-346

License No. NPF-3

Serial No. 1213

November 22, 1985

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz
Operating Reactor Branch No. 4
Division of Licensing
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

On June 17, 1985 Toledo Edison submitted an application for an Amendment to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station Unit No. 1. This application (Serial No. 1159) requested that the action statement concerning the main steam line code safety valves in Technical Specification 3.7.1.1 be revised to require entry into Hot Shutdown rather than Cold Shutdown. As discussed with Mr. George Dick of your staff on November 14, 1985, Toledo Edison is presently re-evaluating this request concerning consistency with the Standard Technical Specifications for B&W PWRs (NUREG-0103, Revision 4). Accordingly, Toledo Edison requests that your office hold in abeyance further processing of this amendment request until Toledo Edison completes its re-evaluation. Toledo Edison will notify your office no later than February 28, 1986 of its position regarding this application.

Very truly yours,

Jo Williams, Jr.

JW:DRW:lah

cc: DB-1 NRC Resident Inspector

8603201131

TOLEDO EDISON

JOE WILLIAMS, JR.
Senior Vice President—Nuclear
(419) 249-2300
(419) 245-5223

Docket No. 50-346

License No. NPF-3

Serial No. 1259

March 20, 1986

Mr. John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stolz:

On June 17, 1985 (Serial No. 1159) Toledo Edison submitted an application for an Amendment to Facility Operating NPF-3 for the Davis-Besse Nuclear Power Station Unit No. 1. The application requested that the action statement concerning the main steam line code safety valves in Technical Specification Section 3.7.1.1 be revised to require entry into hot shutdown rather than cold shutdown.

On November 22, 1985 (Serial No. 1213) Toledo Edison requested that the NRC hold in abeyance further processing of this request while Toledo Edison re-evaluated the request. Toledo Edison has completed this re-evaluation and requests that the NRC continue to process the June 17, 1985 license amendment application.

Yours very truly,

JW:GAB:plf

cc: DB-1 NRC Resident Inspector

Nuclear

GPU Nuclear Corporation
Post Office Box 480
Route 441 South
Middletown, Pennsylvania 17057-019:
717 944-7621
TELEX 84-2386
Writer's Direct Dial Number:

August 11, 1988
C311-88-1088

Mr. D. C. Shelton
Vice President, Nuclear
Toledo Edison
5501 North S.R. 2
Oak Harbor, OH 43449

Dear Mr. Shelton:

At the June 22-23, 1988 BWOG Technical Specification Committee meeting, Dale Wuokko, Davis-Besse Regulatory Affairs Supervisor and I discussed a License Amendment Request that Toledo Edison had submitted dated June 17, 1985 regarding Davis-Besse Technical Specification 3.7.1.1 (Main Steam Safety Valves). This submittal requested that the action for Technical Specification 3.7.1.1 be changed to only require a shutdown to Mode 4 (Hot Shutdown) rather than Mode 5 (Cold Shutdown) since the Technical Specification applicability for operable Main Steam Safety Valves applied only in Modes 1 through 3.

Mr. Wuokko indicated that the NRC Staff planned to deny the request, not due to technical concerns, but rather due to the fact that the requested change deviated from the B&W Standard Technical Specifications. It was therefore proposed that Davis-Besse pursue this change as a lead BWOG plant in order to obtain NRC Staff approval on a generic basis. As Chairman of the BWOG Tech. Spec. Committee, I support Davis-Besse's position on this issue and designate Davis-Besse as the lead BWOG plant. Please keep the BWOG Technical Specification Committee informed as to the NRC's progress in processing this change. Toledo Edison's support of the BWOG Technical Specification Committee in improving the Standard Technical Specifications is appreciated.

Sincerely,

CW Smyth

C. W. Smyth
Chairman,
BWOG Tech. Spec. Committee

CWS/her:1233A

cc: B&WOG

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TOLEDO EDISON

A Centerior Energy Company

DONALD C. SHELTON
Vice President—Nuclear
(419) 249-2300

Docket No. 50-346

License No. NPF-3

Serial No. 1570

August 29, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: Designation of Davis-Besse as Lead BWOG Plant to Revise Action Statement of Main Steam Safety Valves Technical Specification (TAC No. 60103)

Gentlemen:

This letter is being submitted to provide additional information regarding Toledo Edison's License Amendment Request (LAR) of June 17, 1985 (Serial No. 1159). This LAR concerned Tech Spec 3.7.1.1 (Main Steam Safety Valves) and requested that the action for Tech Spec 3.7.1.1 be changed to require a shutdown to Mode 4 (Hot Shutdown) rather than Mode 5 (Cold Shutdown) since the applicability of operable Main Steam Safety Valves only applies in Modes 1 through 3.

On August 11, 1988, the BWOG Technical Specification Committee formally designated the Davis-Besse Nuclear Power Station as the lead BWOG plant in pursuing the Nuclear Regulatory Commission's approval of this change. Therefore, Toledo Edison requests that the NRC Staff process this change as a generic BWOG plant Technical Specification improvement item.

Very truly yours,

MHL/tlt

cc: DB-1 Resident Inspector
A. B. Davis, Regional Administrator
A. W. DeAgazio, NRC/NRR Davis-Besse Project Manager
State of Ohio

EXT-89-03133

Log No. 2903

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

April 25, 1989

RECEIVED

APR 28 1989

TOLEDO EDISON

Docket No. 50-346
Serial No. DB-89-005

Mr. Donald C. Shelton
Vice President, Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Shelton:

SUBJECT: AMENDMENT NO. 132 TO FACILITY OPERATING LICENSE NO. NPF-3
CHANGE IN LIMITING CONDITION FOR MAIN STEAM SAFETY VALVES
(TAC NO. 60103)

The Commission has issued Amendment No. 132 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. This amendment consists of changes to the Appendix A Technical Specifications (TS's) in response to your application dated June 17, 1985 (No. 1159), and letters dated November 22, 1985 (No. 1213), and March 20, 1986 (No. 1159).

The amendment revises Technical Specification 3.7.1.1 concerning the Limiting Condition for Operation for the main steam line safety valves. The change will require the plant to go to Mode 4 (hot shutdown) during valve inoperability, rather than to Mode 5 (cold shutdown) within 12 hours following entry to Mode 3 (hot standby). As requested in your letter dated August 29, 1988, this change has been processed for Davis-Besse as the lead plant of the B&W Owners Group.

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

Sincerely,

Thomas V. Wambach

Thomas V. Wambach, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV,
V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.132to
License No. NPF-3
2. Safety Evaluation

cc: See next page

Mr. Donald C. Shelton
Toledo Edison Company

Davis-Besse Nuclear Power Station
Unit No. 1

cc:
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The Cleveland Electric
Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

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Manager, Nuclear Licensing
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Resident Inspector
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Radiological Health Program
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Attorney General
Department of Attorney
General
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Mr. James W. Harris, Director
(Addressee Only)
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Ohio Environmental Protection Agency
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Columbus, Ohio 43266-0558

President, Board of
County Commissioners of
Ottawa County
Port Clinton, Ohio 43452

State of Ohio
Public Utilities Commission
180 East Broad Street
Columbus, Ohio 43266-0573

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

TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 132
License No. NPF-3

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

   A. The application for amendment by the Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees) dated June 17, 1985 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 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

(a) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 132, are hereby incorporated in the license. The Toledo Edison Company 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 not later than June 9, 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

for Thomas V. Wambach

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V, & Special Projects
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 25, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 132

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following page of the Appendix "A" Technical Specifications with the attached page. The revised page is identified by amendment number and contains vertical lines indicating the area of change. The corresponding overleaf page is also provided to maintain document completeness.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 132 TO FACILITY OPERATING LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter dated June 17, 1985 (No. 1159), the Toledo Edison Company (TE) proposed an amendment to Appendix A Technical Specification (TS) 3.7.1.1. The proposed change would revise the TS Action Statement concerning the Limiting Condition for Operation for the main steam line safety valves. The revision would require the plant to go to Mode 4 (hot shutdown) rather than Mode 5 (cold shutdown) during valve inoperability within 12 hours following entry to Mode 3 (hot standby). TE has proposed this change to eliminate an inconsistency between the Applicability requirement and the Action Statement. In response to the TE request in a letter dated August 29, 1988 (Serial No. 1570), this action has been processed as the lead plant for the B&W Owners Group.

2.0 DISCUSSION

The Main steam overpressure protection system is designed with nine main steam safety valves on each of the two steam (generator) loops. The main steam safety valves are needed to relieve excess steam in the event of various transients to ensure overpressure protection for the plant's secondary side. Pressure relief is required at 1050 psig system design pressure.

The TS 3.7.1.1 Action Statement requires that one or more main steam safety valves inoperable operation in Modes 1, 2, and 3 may proceed if the inoperable valve is restored to operable status or the High Flux Trip Setpoint is reduced (power level reduced) per Table 3.7-1 within 4 hours. Otherwise, the existing TS states that the unit be placed in Mode 3 (hot standby) within the next 6 hours and in Mode 5 (cold shutdown) within the following 30 hours. The proposed revision would, instead, require the plant to enter Mode 4 (hot shutdown) within 12 hours following entry to Mode 3 (hot standby).

T.S. 3.7.1.1 is applicable only in Modes 1, 2, and 3, since the main steam safety valves are required to provide overpressure protection in these modes. Other operating modes (4 and 5) do not exceed a reactor coolant system temperature of 280°F, which corresponds to a saturation pressure of 49.2 psia, substantially below the main steam system design pressure. Therefore, when main steam safety valves are inoperable, operation in Mode 4 is safe since overpressure protection is not required, and it is unnecessary to go to Mode 5. If the operating temperature inadvertently would exceed 280°F in Mode 4, the system would still be protected because of the action of the Reactor Coolant System low temperature overpressure protection features. The staff issued a safety evaluation of the low temperature overpressure protection feature on July 25, 1980. The action of these features would limit pressure to 438 psig.

The proposed 12-hour time requirement to reach Mode 4 is consistent with other TS's that require the plant to go to Mode 4. There is not accident initiating from Mode 4 that would require the operation of the main steam safety valves, and the safety function of TS 3.7.1.1 or the plant is not being degraded by this change. Therefore, the staff finds the proposed changes acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to a surveillance requirement. The staff has determined that the amendment involves 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 this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this amendment meets 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Lynn Kelly

Dated: April 25, 1989

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

May 3, 1989

RECEIVED
MAY 0 9 1989
TOLEDO EDISON

Docket No. 50-346

Mr. Donald C. Shelton
Vice President - Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Shelton:

SUBJECT: CORRECTION TO AMENDMENT NO. 132 TO FACILITY OPERATING LICENSE NO. NPF-3

On April 25, 1989, the Commission issued Amendment No. 132 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment consisted of changes to the Technical Specifications in response to your application dated June 17, 1985.

The amendment revised Technical Specification 3.7.1.1 concerning the Limiting Condition for Operation for the main steam line safety valves. Only page 3/4 7-1 was revised and the marginal lines indicating the area of change were not corrected from the previous amendment No. 117. Page 3/4 7-1, properly marked to reflect the Amendment 132 changes, is enclosed.

Please accept our apologies for any inconvenience this administrative error may have caused you.

Sincerely,

Thomas V. Wambach

Thomas V. Wambach, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
TS Page 3/4 7-1

cc w/enclosure:
See next page

FirstEnergy

Perry Nuclear Power Plant
10 Center Road
P.O. Box 97
Perry, Ohio 44081

Lew W. Myers
Vice President

440-280-5915
Fax: 440-280-8029

Amendment 101

July 13, 1998
PY-CEI/NRR-2298L

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
License Amendment Request Pursuant to 10CFR50.90: Modification of the Safety Setpoint Requirements for the Safety Relief Valves

Ladies and Gentlemen:

Nuclear Regulatory Commission review and approval of a license amendment for the Perry Nuclear Power Plant (PNPP) is requested pursuant to 10CFR50.90. The proposed license amendment request increases the present ±1% tolerance on the safety mode lift setpoint for the safety/relief valves (SRVs) to ±3%. This change has been approved by the NRC staff on a generic basis as documented in NEDC-31753-P-A SER

Attachment 1 provides the Summary, a Description of the Proposed Technical Specification Change, a Safety Analysis, and an Environmental Consideration. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides the annotated Technical Specification page reflecting the proposed change. The annotated Bases pages in Attachment 4 are provided for information only since the Bases are not a formal part of the Technical Specifications. Attachment 5 provides details of the plant specific analysis NEDC-32307P, "Safety Review for Perry Nuclear Power Plant Safety/Relief Valve Setpoint Tolerance Relaxation / Out-of-Service Analyses." This report is considered by General Electric (GE) to be proprietary information and an affidavit from GE to that effect is provided. Pursuant to 10CFR2.790 it is requested that the information contained in Attachment 5 be withheld from public disclosure.

If you have questions or require additional information, please contact Mr. Henry L. Hegrat, Manager - Regulatory Affairs, at (440) 280-5606.

Very truly yours,

Attachments

(Attachment 5 contains 10CFR2.790 information. Upon removal of Attachment 5, the remainder of this package may be disclosed.)

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III
State of Ohio

SUMMARY

The Boiling Water Reactor Owners Group, with the assistance of General Electric, submitted Licensing Topical Report (LTR) NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," for Nuclear Regulatory Commission (NRC) review and approval (Reference 1). The LTR provided justification for the relaxation of safety/relief valve (SRV) safety mode (spring mode) lift setpoint tolerance from ±1% to ±3%. The NRC determined in its corresponding Safety Evaluation Report (NEDC-31753-P-A SER) that it was acceptable for licensees to submit Technical Specification amendment requests to revise lift setting tolerances to ±3% provided that the setpoints, for those SRVs tested, were restored to ±1% (Reference 2). The NRC SER instructed licensees implementing the Technical Specification modifications to provide plant specific analyses confirming the acceptability for revising lift setting tolerances to ±3%. These plant specific analyses and evaluations are summarized below and documented in NEDC-32307P "Safety Review for Perry Nuclear Power Plant Safety/Relief Valve Setpoint Tolerance Relaxation/Out-of-Service Analyses," dated May 1994 (Reference 3 and Attachment 5).

It is estimated that this change will save the Perry Nuclear Power Plant (PNPP) approximately $8.6 million over the remaining life of the plant. In addition, the proposed change will reduce personnel radiation exposure resulting from valve refurbishment and will eliminate additional testing which has no impact on plant safety. We request NRC Staff review and approval of this proposed license amendment by December 31, 1998 in order to support procedure changes necessary to implement this Amendment for the next refueling outage (RFO7) scheduled to start April 10, 1999.

The proposed license amendment request is similar to amendments previously approved by the NRC Staff for the James A. Fitzpatrick Nuclear Power Plant in an SER dated September 28, 1994; for LaSalle County Station, Unit 1 in an SER dated January 3, 1996, and for Grand Gulf Nuclear Station in an SER dated June 12, 1996.

Appropriate Updated Safety Analysis Report (USAR) changes will be completed to describe the change made to increase the SRV safety mode setpoint tolerance to ±3%. It should be noted that no changes are being proposed to the current opening setpoints for the SRVs in their relief mode (pneumatic mode) of operation.

DESCRIPTION OF THE PROPOSED TECHNICAL SPECIFICATION CHANGE

This proposed change increases the present ±1% tolerance on the safety mode lift setpoint for the SRVs to ±3%. Specifically, Surveillance Requirement (SR) 3.4.4.1 is being modified to revise the lift setpoints to reflect:

Setpoint (psig)
1165 ± 34.9
1180 ± 35.4
1190 ± 35.7

The annotated page for the proposed change to SR 3.4.4.1 is provided in Attachment 3.

Additionally, associated Bases changes are included in Attachment 4 for information only, since Bases are not a formal part of the Technical Specifications (Bases changes are processed per the Technical Specification Bases Control Program, Specification 5.5.11).

SAFETY ANALYSIS

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self actuated safety valves. As part of the nuclear pressure relief system, the size and number of SRVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The SRV setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve be set at or below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure (1375 psig - also referred to as the "upset limit"). The overpressure transient evaluation in USAR Chapter 15 is based on these setpoints and includes additional uncertainties of ±1% of the nominal setpoint to account for potential setpoint drift. This provides an additional degree of conservatism.

The LTR NEDC-31753P provided justification for the relaxation of SRV safety mode lift setpoint tolerance from ±1% to ±3% as denoted in SR 3.4.4.1. The NRC determined in its corresponding Safety Evaluation Report that each licensee choosing to implement the SRV setpoint tolerances should provide a plant specific analysis that includes the following:

1) Transient analysis, using NRC approved methods, of the abnormal operational occurrences as described in NEDC-31753P, should be performed utilizing a ±3% setpoint tolerance for the safety mode of the SRVs.

2) Analysis of the design basis overpressurization event using the 3% tolerance limit for the SRV setpoint is required to confirm that the vessel pressure does not exceed the ASME pressure vessel code upset limit.

3) The plant specific analyses described in Items 1 and 2 should assure that the number of SRVs included in the analyses corresponds to the number of valves required to be operable in the Technical Specifications.

4) Re-evaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.), motor-operated valves, and vessel instrumentation and associated piping must be completed, considering the 3% tolerance limit.

5) Evaluation of the ±3% tolerance on any plant specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) should be completed.

6) Evaluation of the effect of the 3% tolerance limit on the containment response during loss of coolant accidents and the hydrodynamic loads on the SRV discharge lines and containment should be completed.

The plant specific analyses and evaluation requested above have been completed and are provided in Attachment 5. Since this is considered a proprietary document, a summary of results is provided below. In addition to the analyses in Attachment 5 (portions of which used PNPP Cycle 5 information), the following summary also notes the applicable results of the current Cycle 7 analyses.

1. TRANSIENT ANALYSIS

Limiting transient events for PNPP Cycle 5 were evaluated (Reference 3) to determine whether the increase in SRV safety (spring) mode setpoint tolerance from ±1% to ±3% would be acceptable. The Cycle 5 evaluation identified the Misoriented Bundle event as the most limiting event for the operating limit minimum critical power ratio (MCPR). The proposed change to the SRV tolerances does not affect the MCPR calculation for the Misoriented Bundle event since there is no SRV actuation in the sequence of events. The evaluation of ΔCPR also included several pressurization transient events (i.e., events in which the SRVs actuate) such as the Generator Load Rejection with no Bypass and the Feedwater Controller Failure. These pressurization events were not affected since credit is taken for SRV operation in the relief mode (pneumatically opened mode of SRV operation), thus, MCPR is not dependent on the SRV safety mode of operation. Also, the safety mode setpoint relaxation is not a factor since the SRVs open after the occurrence of

MCPR during transients. Therefore, the change in the safety mode setpoint does not affect these transient results.

A review of the current Cycle 7 results (USAR 15B.15) identified the Load Reject without Bypass event as the most limiting event for the operating limit MCPR. The proposed change to the SRV tolerances does not affect the MCPR calculation for the Load Reject without Bypass event since there is no impact on MCPR once the reactor scram has occurred. The limiting MCPR occurs prior to the opening of any SRV; and thus the limiting MCPR is not affected.

2. OVERPRESSURE PROTECTION ANALYSIS

The ASME Code requires the peak vessel pressure to remain less than the upset limit of 1375 psig during the limiting overpressure event. For PNPP, this is a Main Steam Isolation Valve (MSIV) closure followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV closure).

An analysis was performed for this event (Reference 3 - Cycle 5 inputs) assuming only 13 of the 19 SRVs installed at PNPP function. The analysis assumed a setpoint tolerance of ±3%, and a 102% power and 105% flow condition with conservative end-of-cycle nuclear dynamic parameters. This analysis ensures the operability requirements of LCO 3.4.4, "Safety/Relief Valves (S/RVs)" are appropriate. For Cycle 5, the peak main steamline pressure was 1264 psig and the peak vessel bottom pressure was 1294 psig.

A review of Cycle 7 results (USAR 15B.5.2.2), which also included the ±3% safety mode setpoint tolerance, identified the peak main steamline pressure as 1258 psig and the peak vessel bottom pressure as 1289 psig. These pressures provide significant margins to the ASME Code upset limit of 1375 psig. Therefore, PNPP satisfies the ASME limit with a ±3% SRV setpoint tolerance using only 13 SRVs. Since the overpressure protection analysis is cycle-specific, the peak vessel pressure is demonstrated every operating cycle to remain below the ASME criteria for overpressure protection.

3. NUMBER OF SRVs USED IN ANALYSIS

The overpressure protection analysis, described in Item 2 above, assumes only the Technical Specification required 13 SRVs are functioning. This is consistent with the Bases for LCO 3.4.4, "Safety/Relief Valves (S/RVs)." The Technical Specification requirements are based on this overpressure protection analysis, not the transient analysis described in Item 1. The plant specific transient analyses, described in Item 1 above for the abnormal operational occurrences, assume that 17 out of 19 SRVs are functioning (USAR 15B.0.1).

4. HIGH PRESSURE SYSTEMS

The impact of the increased setpoint tolerance on the safety functions of the High Pressure Core Spray (HPCS) System, the Reactor Core Isolation Cooling (RCIC) System, and the Standby Liquid Control System (SLCS) was evaluated. The most significant impact is the increased reactor pressure specified for operation. Note that the proposed change does not affect the relief mode setpoints, which remain at:

Setpoint (psig)	
1103 ± 15	(1 valve)
1113 ± 15	(9 valves)
1123 ± 15	(9 valves)

High Pressure Core Spray System

The HPCS System is designed to deliver water to the reactor vessel at ≥ 517 gpm, with the reactor vessel pressure 1177 psig (1165 + 11.6 psig) above the pressure at the source of suction. The increase in SRV setpoint tolerance increases the maximum reactor pressure for HPCS System injection to 1200 psig (1165 + 34.9 psig). A review of the pump performance curves indicates that the HPCS System has sufficient margin to deliver flow in excess of 517 gpm at 1200 psig. The HPCS pump design has a discharge rating of 1575 psig. This pressure is well above the pressures which may result from SRV safety mode setpoint tolerance relaxation. Therefore, the increase in SRV setpoint tolerance has been determined to be acceptable for the HPCS System.

Reactor Core Isolation Cooling System

The design requirements for the RCIC System, including the RCIC turbine, is to deliver 700 gpm in the reactor pressure range of 150 to 1177 psig. The increase in SRV setpoint mode tolerance requires the RCIC System to deliver 700 gpm in the reactor pressure range of 150 to 1200 psig. A review of the pump performance curves indicates that the RCIC System has sufficient margin to deliver flow in excess of 725 gpm at 1200 psig. The RCIC pump design has a discharge rating of 1575 psig. This pressure is well above the pressures which may result from SRV safety mode setpoint tolerance relaxation. Therefore, the increase in SRV setpoint tolerance has been determined to be acceptable for the RCIC System.

A review of the RCIC turbine performance curves indicates that the turbine has the capacity to develop the horsepower and speed required by the pump for the increased pressure conditions. The turbine speed must be raised from 4550 rpm to 4600 rpm. This is below the maximum permitted turbine shaft speed of approximately 5000 rpm. The increased turbine rated speed reduces the overspeed trip margin from 125% to 122.3%. However, this reduction in margin was found to be acceptable in accordance with the GE Service Information Letter No. 377. The steam flow increase to drive the turbine to a higher speed is less than 2%. This steam flow increase and a small increase in turbine exhaust pressure are not expected to have any adverse impact to the design life and performance of the turbine system. The design pressure of the RCIC System steam supply lines and turbine was based on the reactor design pressure of 1250 psig. This pressure is above the pressures which may result from SRV safety mode setpoint tolerance relaxation. Therefore, the increase in SRV setpoint tolerance has been determined to be acceptable for the RCIC System.

Standby Liquid Control System

The SLCS was evaluated in Reference 3 based on the SRV safety mode settings. However, SLCS operation is not affected by the SRV setpoint tolerance increase. The pressure used for system performance is based on the SRV relief settings of the system, not the SRV safety settings. This proposed change does not affect the relief settings of the SRVs.

Other Components (MOVs, Associated Piping)

The minimum design pressure of the piping, valves and components which are part of the primary reactor coolant pressure boundary is 1250 psig. Since the design pressure is higher than the lowest opening pressure of the SRV safety mode at 1200 psig, an increase in SRV setpoint tolerance has been determined to be acceptable.

5. ALTERNATE OPERATING MODES

The alternate operating modes, including Maximum Extended Operating Domain (MEOD), Increased Core Flow Region, and Single Loop Operation (SLO) were considered in determining the most restrictive analytical conditions (i.e., the most limiting operating mode) for performing the analyses associated with this proposed Technical Specification change (see Items 1 and 2, above).

6. LOSS OF COOLANT ACCIDENT (LOCA) PERFORMANCE

LOCA

The LOCA analysis was reviewed to determine the effect of an increase in SRV setpoint tolerance on ECCS performance. The ECCS is designed to maintain fuel integrity, during postulated LOCAs, below 10CFR50.46 limits. A change in SRV opening pressure can only affect the containment pressure response for LOCAs in which SRV actuations occur. No SRV actuation occurs for large pipe breaks inside the containment because the vessel depressurizes through the break. Therefore, an increase in SRV opening pressure will not affect the results of the design basis accident (DBA) LOCA evaluated in the USAR. For a double-ended guillotine break of one main steamline outside the containment, the reactor vessel is isolated upon closure of the MSIVs. The peak pressure of this accident is bounded by the overpressure protection event. In addition, 10CFR50.46 calculations of peak clad temperatures for this event and for small-break LOCAs are insensitive to SRV actuation. For small break LOCAs, the SRVs are armed with low-low set logic (LLS) allowing the relief mode of 6 SRVs to relieve reactor pressure to below 1000 psig. Once the logic is initiated, the opening and closing setpoints of these SRVs are automatically reset to lower values by the LLS logic. This logic is not affected by the setpoint tolerance change since it acts on the relief mode of SRV actuation and not on the safety mode of operation. Consequently, since PNPP is a LLS plant, the peak pressure effect from the SRV setpoint relaxation on a small break LOCA event is negligible.

Containment Response

The most limiting LOCA event in terms of peak containment pressure, temperature and peak suppression pool temperature is a DBA LOCA. Relaxation of the SRV setpoint tolerance has no effect on this limiting event because the vessel depressurizes without any SRV actuation.

LOCA-Related Hydrodynamic Loads

The LOCA hydrodynamic loads, such as pool swell, condensation oscillation and chugging, are all dependent on peak containment responses during a DBA LOCA and therefore are not affected by the increase in the SRV safety mode setpoint tolerances.

SRV Discharged Loads

Steam discharged from the SRVs is routed through the SRV discharge lines and through the quencher into the suppression pool. Actuation of SRVs introduces high pressure steam which quickly pressurizes the discharge piping resulting in forced expulsion of the water as well as air initially in the piping. The SRV loads resulting from SRV actuation includes thrust loads, air-clearing loads, reaction loads, and air bubble loads impacting SRV piping, piping anchors, quenchers, and submerged structures (USAR Appendix 3B). An increase of the SRV setpoint tolerance will result in a corresponding increase in discharge flow rate into the discharge piping. At 1%, the maximum pressure setting can be as high as 1202 psig. At 3%, the maximum pressure setting can be as high as 1226 psig. The maximum increase is only 24 psig or about 2 %. The effect of SRV discharged loads from the SRV safety mode setting tolerance increase can be evaluated as follows:

- SRV Discharge Piping to the First Anchor - The original loads for this portion of piping were generically derived for all BWR/6 plants and were based on worst case input conditions. The resultant SRV load was developed using an SRV flow rate input higher than the maximum expected SRV flow rate (with 3% setpoint tolerance) for PNPP. As indicated in USAR Table 3.9.3 (k), the resultant stresses for this portion of SRV piping show large safety margins in excess of 40%.

- Quencher Loads - PNPP uses the standard GE X-quencher design. The generic loads defined for the original quencher loading design were very conservative and therefore were determined to be unaffected by the increase in SRV setpoint tolerance.

- Submerged Structure and Pool Boundary Loads - The loads on the submerged structures are based on the peak air bubble pressures determined with the generic X-quencher methodology. The correlation which was used to determine the increase in the SRV generic peak bubble pressure due to the increase in the SRV discharge flow rate was determined to be very conservative. Therefore there is no effect on the submerged structures load definition.

ATWS Mitigation Capability

An Anticipated Transient Without Scram (ATWS) is a beyond design basis event. The potential impact of the SRV tolerance setpoint relaxation on ATWS performance is the compliance to vessel overpressure criteria of 1500 psig. The limiting event for this ATWS condition is the MSIV closure transient (assumes that a reactor scram does not occur on a reactor protection system signal). For this event, the initial reduction of power occurs from the ATWS high dome pressure recirculation pump trip, accompanied by the boron injection from the SLCS. This leads to an eventual reactor shutdown. During this transient, SRVs actuated accordingly. An analysis was performed with the reactor at 100% power and 100% recirculation flow. Assuming 2 SRVs failed to open on demand, the peak reactor vessel bottom pressure was calculated to be 1344 psig. This is significantly less than the acceptance limit, which is the ASME service level C (Emergency) value of 1500 psig. Therefore, the setpoint tolerance relaxation on the safety mode lift setpoint does not adversely impact the results of any ATWS event.

7. CONCLUSION

The analyses and evaluations support the relaxation of the as-found SRV safety mode lift setpoint tolerance from the current ±1% to ±3%. The analyses and evaluations summarized above have no significant safety impact on ECCS/LOCA performance, high pressure system performance, containment structural integrity, and ATWS analysis results. The analyses examined cycle dependent safety concerns, such as vessel overpressure margin and thermal limits, and demonstrated that the SRV safety mode tolerance combined with 2 SRVs out-of-service has no adverse impact upon plant safety for transient events. In addition, there is no adverse impact upon plant safety with 6 SRVs out-of-service for the vessel overpressure analysis. Future cycle-specific reload licensing evaluations verify continued applicability of the results from this analysis.

8. REFERENCES

1. NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," GE Nuclear Energy, dated February 1990.

2. NRC letter (NEDC-31753-P-A SER) from A.S. Thadani to C.L. Tully, Chairperson, BWR Owners' Group, "Acceptance for Referencing of Licensing Topical Report NEDC-31753P," dated March 8, 1993.

3. NEDC-32307P, "Safety Review for Perry Nuclear Power Plant Safety/Relief Valve Setpoint Tolerance Relaxation / Out-of-Service Analyses," GE Nuclear Energy, dated May 1994.

ENVIRONMENTAL CONSIDERATION

The proposed Technical Specification change request was evaluated against the criteria of 10CFR51.22 for environmental considerations. The proposed change does not significantly increase individual or cumulative occupational radiation exposures, does not significantly change the types or significantly increase the amount of effluents that may be released off-site and, as discussed in Attachment 2, does not involve a significant hazards consideration. Based on the foregoing, it has been concluded that the proposed Technical Specification change meets the criteria given in 10CFR51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.

COMMITMENTS WITHIN THIS LETTER

The following table identifies those actions which are considered to be regulatory commitments. Any other actions discussed in this document represent current or planned actions and are described for the NRC's information. Please notify the Manager - Regulatory Affairs at the Perry Nuclear Power Plant of any questions regarding this document or any associated regulatory commitments.

Commitments
Appropriate Updated Safety Analysis Report (USAR) changes will be completed to describe the change made to increase the SRV safety mode setpoint tolerance to ±3%.

SIGNIFICANT HAZARDS CONSIDERATION

The standards used to arrive at a determination that a request for amendment involves no significant hazards considerations are included in the Commission's Regulation, 10CFR50.92, which states that the operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed amendment has been reviewed with respect to these three factors and it has been determined that the proposed change does not involve a significant hazard because:

(1) The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change allows an increase in the as-found safety relief valve (SRV) safety mode setpoint tolerance, determined by test after the valves have been removed from service, from ±1% to ±3%. The proposed change does not alter the Technical Specification requirements on the nominal SRV safety mode lift setpoints, the SRV relief mode setpoints, the required frequency for the SRV lift setpoint tests, or the number of SRVs required to be operable. This change does not involve physical changes to the SRVs, nor does it change the operating characteristics or safety function of the SRVs.

Consistent with current requirements, this change continues to require that the SRVs be adjusted to within ±1% of their nominal lift setpoints following testing. This change does not change the behavior and operation of any SRV and therefore has no significant impact to reactor operation. It also has no significant impact on response to any perturbation of reactor operation including transients and accidents previously analyzed in the Updated Safety Analysis Report. In addition, this change does not change SRV actuation. Therefore, this change will not increase the probability of an accident previously evaluated.

Generic considerations related to the change in setpoint tolerance were addressed in NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," and were reviewed and approved by the NRC. The plant specific evaluations, required by the NRC's Safety Evaluation for NEDC-31753P and performed to support this proposed change, are contained in NEDC-32307P, "Safety Review for PNPP Safety/Relief Valve Setpoint Tolerance Relaxation / Out-of-Service Analyses," dated May 1994. These analyses and evaluations show that there is adequate margin to the design core thermal limits and to the reactor vessel pressure limits using a ±3% SRV setpoint tolerance. They also show that operation of the high pressure injection systems will not be adversely affected; and the containment response from a loss of coolant accident will be acceptable.

Therefore, this change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

(2) The proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to allow an increase in the SRV safety mode setpoint tolerance from ±1% to ±3% does not alter the nominal SRV lift setpoints or the number of SRVs required to be operable. This change does not involve physical changes to the SRVs, nor does it change the operating characteristics or the safety function of the SRVs. The proposed change does not involve a physical alteration of the plant. No new or different equipment is being installed. The proposed change does not impact core reactivity nor the manipulation of fuel bundles. There is no alteration to the parameters within which the plant is normally operated. As a result no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) The proposed change will not involve a significant reduction in the margin of safety.

The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above considerations, it is concluded that a significant hazard would not be introduced as a result of this proposed change. Also, since NRC approval of this change must be obtained prior to implementation, no unreviewed safety question can exist.

TECHNICAL SPECIFICATION PAGE REVISIONS

TECHNICAL SPECIFICATION BASES REVISIONS

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 Safety/Relief Valves (S/RVs)

INFORMATION ONLY

BASES

BACKGROUND

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 1) requires the Reactor Pressure Vessel be protected from overpressure during upset conditions by self actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety/relief valves (S/RVs) are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each S/RV discharges steam through a discharge line and quencher to a location below the minimum water level in the suppression pool. With one or more S/RVs stuck open, operators will close the S/RV, thus minimizing the increase in suppression pool water temperature.

The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the direct action of the steam pressure in the main steam lines will act against a spring loaded disk that will pop open when the valve inlet pressure exceeds the spring force. In the relief mode (or power actuated mode of operation), a pneumatic operator and mechanical linkage assembly are used to open the valve by overcoming the spring force, even with the valve inlet pressure equal to 0 psig. The pneumatic operator is arranged so that its malfunction will not prevent the valve disk from lifting if steam inlet pressure reaches the spring lift set pressures. In the relief mode, valves may be opened manually or automatically at the selected preset pressure. Six of the S/RVs providing the relief function also provide the low-low set relief function specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves." Eight of the S/RVs that provide the relief function are part of the Automatic Depressurization System specified in LCO 3.5.1, "ECCS—Operating." The instrumentation associated with the relief valve function and low-low set function is discussed in the Bases for LCO 3.3.6.4, "Relief and Low-Low Set (LLS) Instrumentation."

(continued)

BACKGROUND (continued)	and instrumentation for the ADS function is discussed in LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation."
APPLICABLE SAFETY ANALYSES	The overpressure protection system must accommodate the most severe pressure transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs) followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 2). For the purpose of the analyses, the 13 safety valves with the highest setpoints were assumed to be operational. Therefore, by requiring six S/RVs to be OPERABLE in the relief mode and seven in the safety mode, the accident analyses assumptions are adequately met. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the design basis event. Reference 3 discusses additional events that are expected to actuate the S/RVs. From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. S/RVs satisfy Criterion 3 of the NRC Policy Statement.
LCO	The safety function of seven S/RVs is required to be OPERABLE in the safety mode, and an additional six S/RVs (other than the seven S/RVs that satisfy the safety function) must be OPERABLE in the relief mode. The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure. In Reference 2, an evaluation was performed to establish the parametric relationship between the peak vessel pressure and the number of OPERABLE S/RVs. The results show that with a minimum of seven S/RVs in the safety mode and six S/RVs in the relief mode OPERABLE, the ASME Code limit of 1375 psig is not exceeded. The S/RV setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve be set at or

(continued)

BASES

LCO (continued)

below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for conditions. The transient evaluations in Reference 3 are based on these setpoints, but also include the additional uncertainties of ±1% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism. Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

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APPLICABILITY

In MODES 1, 2, and 3, the specified number of S/RVs must be OPERABLE since there may be considerable energy in the reactor core and the limiting design basis transients are assumed to occur. The S/RVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the heat.

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If one or more required S/RVs are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This Surveillance demonstrates that the required S/RVs will open at the pressures assumed in the safety analysis of Reference 2. The demonstration of the S/RV safety function lift settings must be performed during shutdown, since this is a bench test, and in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

The Frequency was selected because this Surveillance must be performed during shutdown conditions and is based on the time between refuelings.

INSERT B 3.4-21A

SR 3.4.4.2

The required relief function S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions i.e., solenoids of the automatic relief function operate as designed when initiated either by an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.4.4 overlaps this SR to provide complete testing of the safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation. This prevents an RPV pressure blowdown.

SR 3.4.4.3

A manual actuation of each required S/RV is performed to verify that the valve is functioning properly and that no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, by the S/RV discharge pipe pressure switch, or any other method suitable to verify steam flow (e.g., tailpipe temperature.)

(continued)

INSERT B 3.4-21A

The safety lift setpoints will still be set within a tolerance of ± 1%, but the setpoints will be tested to within ± 3% to determine acceptance or failure of the as-found valve lift setpoint (Reference 4).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.3 (continued)

Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is the pressure recommended by the valve manufacturer. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. SR 3.4.4.2 and the LOGIC SYSTEM FUNCTIONAL TEST performed in SR 3.3.6.4.4 overlap this surveillance to provide complete testing of the assumed safety function. If the valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The 18 months on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 18 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 1). Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Sections III and XI.

2. USAR, Chapter 15, Appendix 15B.

3. USAR, Section 15.

4. NRC Safety Evaluation to NEDC-31753P, March 8, 1993.

Attachment 5
PY-CEI/NRR-2298L

SAFETY REVIEW FOR PERRY NUCLEAR POWER PLANT SAFETY/RELIEF VALVE SETPOINT TOLERANCE RELAXATION / OUT-OF-SERVICE ANALYSES

(NEDC-32307P)

PROPRIETARY

Amendment 101

FirstEnergy

Perry Nuclear Power Plant
10 Center Road
Perry, Ohio 44081

Lew W. Myers
Vice President

440-280-5915
Fax: 440-280-8029

November 23, 1998
PY-CEI/NRR-2332L

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
Response to Request for Additional Information Regarding
Safety Relief Valve Setpoint Tolerance (TAC No. MA2290)

Ladies and Gentlemen:

In a letter dated September 16, 1998, the NRC staff issued a Request for Additional Information (RAI) concerning a request for revision of the safety/relief valve setpoint tolerance for the Perry Nuclear Power Plant (PNPP). The change to the setpoint tolerance was requested in a letter dated July 13, 1998 (PY-CEI/NRR-2298L).

Attachment 1 herein provides the response to three (3) NRC RAI questions and two supplemental information issues not specifically addressed in the RAI, but discussed with the NRC staff. The RAI responses and supplemental clarifications neither modify the proposed Technical Specification change nor impact the proposed Significant Hazards Consideration provided in the original submittal.

If you have questions or require additional information, please contact Mr. Henry L. Hegrat, Manager-Regulatory Affairs, at (440) 280-5606.

Very truly yours,

Lew W Myers

Attachment

cc: NRC Region III
NRC Resident Inspector
NRC Project Manager
State of Ohio

PNPP RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION ON SAFETY/RELIEF VALVE SETPOINT TOLERANCE

Question 1

"Uncertainties of analysis parameters should be accounted for in the safety analyses used to justify new limits in the plant Technical Specifications (TSs). Provide a discussion of the Safety/Relief Valve (SRV) setpoint testing instrument accuracy and how this source of uncertainty is accounted for in the licensee's safety analysis associated with the proposed TS SRV setpoint tolerance."

Response to Question 1

The setpoint testing instrument accuracy is accounted for in the testing of the SRVs, not in the safety analysis. The Perry Nuclear Power Plant (PNPP) specifies minimum test instrument loop accuracy within the vendor contract for SRV benchtesting and refurbishment. The vendor currently used by PNPP to conduct these activities utilizes a test bench that has a stacked test instrument loop accuracy of 0.15% of the indicated (measured) set pressure. Test instrument loop accuracy is accounted for in all "as-found" and "as-left" benchtest results. The vendor is required to notify PNPP whenever the as-tested results are outside the setpoint tolerance taking into account the stacked test instrument loop accuracy.

Question 2

"Provide verification that the functional capability of all safety-related motor operated valves has been evaluated for the larger differential pressure loads resulting from the increased SRV setpoint tolerance in accordance with the Generic Letters 89-10 and 96-05 programs. In addition, provide verification that the functional capability of all safety-related air-operated and hydraulically-operated valves has not been adversely affected by the increased SRV tolerance."

Response to Question 2

A maximum expected differential pressure (MEDP) calculation is performed for valves in the Generic Letters (GL) 89-10 and 96-05 program to establish a maximum differential pressure expected during various operational conditions. MEDP calculations were prepared for MOVs included in the GL program, and were developed based on industry guidance developed generically for all BWRs, and considering PNPP plant specific requirements.

For PNPP, dynamic testing of MOVs is done at the highest differential pressure achievable under normal operational configurations for selected valves in established valve groups. Therefore, dynamic testing requirements are unaffected by the SRV safety setting tolerance increase. However, MOV operator settings for static testing are based on the calculated MEDP values (as one of the input parameters for determining required settings). Adequacy of MOV settings was assessed by evaluating the adequacy of the MEDP calculation assumptions and resulting MEDP values established for those valves potentially affected by the increase in SRV safety setpoint tolerance.

Plants defined in NEDC-31753P as Group 3 (i.e., BWR 5/6), have SRVs with two (dual) modes of operation (relief and safety), and credit the safety-grade externally powered relief mode in the analysis of abnormal operational occurrences (AOOs), the ASME overpressurization analysis, and the Anticipated Transient Without Scram (ATWS) events. The relief mode is also utilized in the calculation of the MEDP for the valves in the Generic Letters (GL) 89-10 and 96-05 program. For the SRVs operating in the relief mode there are three groups of relief setpoints. The highest setpoint group opens at 1123 psig ±15 psig. Thus, the highest relief mode pressure is 1138 psig, and all SRVs credited in the accident and transient analyses for relief mode operation would open if reactor pressure exceeded this value. This is below the lowest SRV safety mode setpoint. Therefore, the SRV safety mode setpoint tolerance increase to 3% does not affect the MEDP for these valves.

In addition, there are no safety-related air-operated and hydraulically-operated valves whose functional capability will be adversely affected by the increased SRV tolerance.

Question 3

"The NRC Safety Evaluation for topical report NEDC-31753P, dated March 8, 1993, stated that in order to increase the SRV setpoint tolerance from ±1% to ±3%, half of the SRVs should be tested every 18 months and all SRVs should be tested within 40 months. The licensee's letter dated July 13, 1998 proposes to continue testing the SRVs less frequently in accordance with the plant inservice testing program. Provide the basis for testing the SRVs less frequently than that approved for the above topical report while increasing the SRV setpoint tolerance to ±3%, or revise the proposed TS accordingly."

Response to Question 3

The current licensing basis for PNPP, as reflected in the Updated Safety Analysis Report, Section 5.2.2.10, "Testing and Inspection", is that at least half of its SRV population is removed and tested each Refuel Outage. In addition, the benchtest interval for any individual valve shall not exceed 5 years. This is also reflected in the original NRC Safety Evaluation Report Related To The Operation Of The Perry Nuclear Power Plant, dated May 1982, Section 5.2.3. No changes to this schedule are proposed by the pending license amendment request.

ADDITIONAL DISCUSSIONS

In addition to the above information provided in the Request for Additional Information, the following is provided as supplemental information to clarify two (2) items specifically discussed in NEDC-31753P and NEDC-32307P.

Item 1

The NRC Safety Evaluation Report for topical report NEDC-31753P, dated March 8, 1993, states that "Re-evaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.), motor operated valves, and vessel instrumentation and associated piping must be completed, considering the 3% tolerance limit." The PNPP submittal dated July 13, 1998 (PY-CEI/NRR-2298L), did not specifically address vessel instrumentation or the piping connected to that instrumentation.

Instruments which could be affected by the possible increase in pressure resulting from the proposed change were evaluated with respect to effects on pressure boundary integrity, instrument calibration, and instrument scaling calculations and instrument setpoint/uncertainty calculations (as applicable). Instruments in high pressure systems such as the Control Rod Drive and Standby Liquid Control Systems were excluded because the systems are designed to operate at pressures higher than that resulting from the SRV tolerance relaxation.

A review of vendor information for each instrument indicated that the increased pressure is within the pressure boundary design limit. Calibration information for the instruments was reviewed and the calibration range of all instruments is adequate considering a potential higher reactor pressure of 1200 psig.

Instrument scaling calculations use normal operating pressures as an input, rather than anticipated maximum pressures and are, therefore, not affected by this proposed change. Additionally, the reactor pressure values used in determination of static pressure effects and overpressure effects in instrument setpoint and uncertainty calculations bound the pressure resulting from the proposed SRV setpoint tolerance relaxation.

For PNPP, piping connected to the reactor coolant pressure boundary (RCPB) is designed for pressures equal to or greater than rated reactor vessel design pressure of 1250 psig. Instrument piping/tubing class is determined by the process pipe class. Additionally, instrument piping connected directly to the reactor vessel has a minimum design pressure rating of 1250 psig. All of this piping is protected from overpressurization by the SRVs which satisfy ASME Code requirements for overpressure protection for the reactor vessel and connected piping. Pressure transients associated with upset and faulted conditions analyzed in the USAR are bounded by core reload analyses which utilize a +3% tolerance for SRV safety mode operation in evaluating maximum overpressurization scenarios. Therefore, RCPB piping, including the instrument piping within the RCPB, has adequate design margin for overpressure protection.

Therefore, the proposed change in SRV setpoint drift tolerance has no impact on plant instrumentation and instrument piping/tubing.

Item 2

The topical report NEDC-32307P, Section 6.2.6, submitted with the letter dated July 13, 1998 (L.W. Myers to the Nuclear Regulatory Commission, "License Amendment Request Pursuant to 10CFR50.90: Modification of the Safety Setpoint Requirements for the Safety Relief Valves") recommended that the RCIC steam supply line isolation differential pressure set point value be re-evaluated.

RCIC steam flow is monitored by flow instrumentation for the purpose of isolating steam to the turbine if an excessive steam flow rate, indicative of a line break, occurs. The instrumentation setpoint is equivalent to 300% of steam flow at the design condition of 1192 psia and 700 gpm pump flow. Although operation beyond this design condition requires slightly increased steam flow, the margin to isolation on high steam flow is not significantly impacted. The existing high steam flow setpoint is slightly conservative (lower) than one calculated based on 300% of steady state steam flow at 1215 psia. Therefore, the evaluation concluded that the existing RCIC high steam flow setpoint would not be revised.

Commitments

There are no regulatory commitments contained in this letter.

cc:

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 101
License No. NPF-58

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

   A. The application for amendment by the FirstEnergy Nuclear Operating Company (the licensee, formerly The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, and Toledo Edison Company) dated July 13, 1998, as supplemented by submittal dated November 23, 1998, 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 reouirements 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 2.C.(2) of Facility Operating License No. NPF-58 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 101 are hereby incorporated into this license. The FirstEnergy Nuclear Operating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett

Douglas V. Pickett, Senior Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 3, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 101

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3.4-10	3.4-10

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101TO FACILITY OPERATING LICENSE NO. NPF-58

FIRSTENERGY NUCLEAR OPERATING COMPANY

PERRY NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter dated July 13, 1998 (Ref.1), the FirstEnergy Nuclear Operating Company, the licensee for Perry, submitted proposed changes to Technical Specification (TS) section 3.4.4, "Safety/Relief Valves (SRVs)." The licensee submitted additional information in a letter dated November 23, 1998 (Ref. 2). The changes would allow the licensee to increase the allowable safety/relief valve (SRV) as-found setpoint tolerance from ±1% to ±3%.

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original Federal Register notice.

2.0 BACKGROUND

10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor coolant system design" states that "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

The proposed change does not alter the SRV safety lift setpoints, relief setpoints, the SRV lift setpoint test frequency, or the number of SRVs required to be operable. Also, the proposed change requires the as-left safety valve function settings to be within ±1% of the specified nominal lift setpoints prior to installation before testing. The staff has previously granted approval to individual BWRs to increase the as-found SRV tolerance to three percent. The basis for the approval was a staff safety evaluation report (SER) for a licensing topical report (LTR NEDC-31753P) evaluating the setpoint tolerance increase. The staff SER (Ref. 3) included six conditions which must be addressed on a plant-specific basis for licensees applying for the increased SRV setpoint tolerance:

(a) Transient analysis of all abnormal operational occurrences as described in NEDC-31753P (Ref. 4), should be performed utilizing a ±3% tolerance for the safety mode of spring safety valves (SSVs) and SRVs. In addition, the standard reload methodology (or other method approved by the staff) should be used for this analysis.

(b) Analysis of the design basis over-pressurization event using the 3% tolerance limit is required to confirm that the vessel pressure does not exceed the ASME pressure vessel code upset limit.

(c) The plant-specific analysis described in items (a) and (b) should assure that the number of SSVs, SRVs, and relief valves (RVs) included in the analyses correspond to the number of valves required to be operable in the technical specification.

(d) Reevaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc,), motor-operated valves, and vessel instrumentation and associated piping must be completed, considering the 3% tolerance limit.

(e) Evaluation of the ±3% tolerance on any plant-specific operating modes (e.g., increased core flow, extended operating domain, etc.) should be completed.

(f) Evaluation of the effect of the 3% tolerance limit on the containment response during loss of coolant accidents and the hydrodynamic loads on the SRV discharge lines and containment should be completed.

3.0 EVALUATION

The safety objective of the SRVs is to prevent over-pressurization of the nuclear system. This protects the nuclear system process barrier from failure which could result in the uncontrolled release of fission products. The pressure relief system at Perry includes nineteen SRVs, arranged into three setpoint groupings: one group of SRVs (8) set at 1165 psig, a second group of SRVs (6) set at 1180 psig, and a third group of SRVs (5) set at 1190 psig. Existing TS provides a ±1% as-found tolerance and ±1% as-left setpoint tolerance. The proposed modifications would provide a ±3 % as-found tolerance and ±1% as-left setpoint tolerance. The licensee's submittal was evaluated against the generic SER described above.

3.1 Transient Analysis / Reload Methodology

The licensee must consider the impact of the tolerance increase on abnormal operational transients (AOTs). For Perry, analysis (cycle 7 reload analysis) of AOTs has been conducted utilizing the 3% tolerance and with 17 of the total 19 SRVs in service. All future reload analyses are expected to assume the 3% tolerance. The transient which generates the limiting decrease in a critical power ratio is the load rejection without turbine bypass event. The analysis showed that the thermal limits of the limiting transient would not be affected by the relaxation of SRV setpoint tolerance. Further, other transient events remain non-limiting and bounded by the above event. The staff finds the licensee's analysis acceptable because it was performed using a methodology previously approved by the NRC (Ref. 5).

3.2. Analysis of the Design Basis Overpressurization Event

The licensee is required to reevaluate the limiting design-basis pressurization transient using the 3% tolerance limit to confirm that the vessel pressure does not exceed the American Society of Mechanical Engineers (ASME) pressure vessel code upset limit. The ASME Boiler and Pressure Vessel Code Section III permits pressure transients up to 10% over design pressure (110% x 1250 psig = 1375 psig). The limiting pressurization AOT analyzed is a main steam isolation valve (MSIV) closure event occurring at the end of full power life without credit for a reactor trip on MSIV position sensing. The licensee analyzed (Ref.1) the MSIV closure event using the staff-approved model ODYN with the 3% tolerance and calculated the maximum vessel pressure to be 1289 psig. This is within the 1375 psig ASME limit, and is acceptable to the staff.

3.3. TS Operability Statement for SRVs

The licensee has stated that plant-specific overpressure analyses (Ref.1) have been conducted with the number of SRVs included in the analyses corresponding to the number of valves required to be operable in TS. The analysis took credit only for 13 of the 19 SRVs required by the TS. This is acceptable to the staff.

The surveillance frequency of the SRVs is specified in the plant TS to be in accordance with the plant in-service testing (IST) program. The IST program is required to meet the ASME Code which specifies that the SRVs must be tested at least every 5 years. However, the licensee stated that the current licensing basis for Perry is that at least half of the SRV population is removed and tested each refueling outage. This test frequency is sufficient to meet the test frequency specified in the staff SER (Ref. 3) for LTR NEDC-31753P, and is acceptable.

3.4. Reevaluation of the Performance of High Pressure Systems

The licensee must also reevaluate performance of high pressure systems (pump capacity, discharge pressure, etc.), considering the 3% tolerance limit. Perry has three systems which are required to inject to the vessel at high pressure conditions: high pressure core spray (HPCS), reactor core isolation cooling (RCIC), and standby liquid control system (SLCS). The most significant impact is the increased reactor pressure specified for system operation. The systems' performances were evaluated for the new reactor pressure of 1200 psig from 1177 psig. The HPCS system was determined to have the capability to inject its design flow of 517 gpm to the vessel at the new maximum pressure of 1200 psig without any changes. The RCIC turbine maximum steam flow rate is increased from 34,200 lbm/hr to 34,800 lbm/hr. The RCIC turbine/pump maximum speed is increased from 4550 rpm to 4600 rpm in order for the RCIC system to perform at the new maximum reactor operating pressure. The increased speed reduces the over-speed margin from 125% to 122.3%. This reduction in margin is acceptable due to the system modifications to the turbine start feature. The SLCS system was determined to have the capability to inject boron into the vessel at its design flow rate at the higher reactor pressures.

3.5. Evaluation of Motor-Operated Valves and Piping

In support of the SRV tolerance increase from ±1% to ±3%, the licensee stated that a maximum expected differential pressure calculation was performed for all valves in the Generic Letters (GL) 89-10 and 96-05 program for various operational conditions. The licensee determined that for dynamic and static testing of the MOVs, there is no effect on the maximum expected differential pressures resulting from the SRV safety setting tolerance increase. The staff finds that meeting the requirements of the GLs 89-10 and a 96-05 program is sufficient regarding required operational capability of the MOVs. The licensee further stated that there are no safety-related, air-operated, or hydraulically-operated valves whose functional capability would be adversely affected by the increased SRV tolerance, which is acceptable.

An increase in SRV setpoint tolerance involves a potential increase in SRV discharge hydrodynamic loads on the SRV discharge piping and quencher, the submerged structures, and the suppression pool boundary. The licensee reviewed the load increase to determine if sufficient conservatism and margins are available in the currently defined SRV loads. As a result, the licensee determined that the increase in SRV opening setpoint pressure would not adversely impact the current design-basis SRV hydrodynamic load analysis results.

The licensee also evaluated the effects of the high pressures associated with the increased setpoint tolerance on the instrumentation and piping for the systems. The licensee determined that no changes to instrumentation will be required. The licensee also determined that the impact of the higher pressure on system piping and other components was negligible.

The staff believes that the licensee has performed the appropriate analysis to determine any adverse impact of the proposed changes on motor operated valves and piping. The staff has reviewed the methodology used by the licensee for the above evaluations and results and concludes that it is acceptable.

3.6. Alternate Operating Modes

The licensee must also evaluate the increased tolerance on any plant-specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) The analyses included evaluations for the currently approved operating domains: Maximum Extended Operating Domain (MEOD), Increased Core Flow and Single Loop Operation. The analyses were found acceptable by the staff.

3.7. Containment Response / Hydrodynamic Loads

As previously described, the pressure relief system at Perry includes 19 SRVs, arranged into three setpoint groupings: one group of SRVs (8) set at 1165 psig, a second group of SRVs (6) set at 1180 psig, and a third group of SRVs (5) set at 1190 psig. The tolerance level of ±1% was small enough to have maintained three distinctive bands. The inference being that during transient conditions, the SRVs would be expected to open in discrete groupings, or bands, as opposed to all 19 SRVs opening at once. This effect can be seen by looking at the minimum and maximum values of acceptable lifting SRV pressures. The following array provides this information for each of the three distinctive setpoints:

1165 psig with an acceptable range of 1153.3 psig to 1176.6 psig
1180 psig with an acceptable range of 1168.2 psig to 1191.8 psig
1190 psig with an acceptable range of 1178.1 psig to 1201.9 psig

As seen above, the existing SRV setpoint tolerances result in minimal overlap between the three setpoint settings. While banding is generally retained for the setpoint tolerance of ±1%, increasing the setpoint tolerance to ±3% allows for more significant overlap of tolerances. A similar presentation including ±3% tolerances produces the following results:

1165 psig with an acceptable range of 1130.1 psig to 1199.9 psig
1180 psig with an acceptable range of 1144.6 psig to 1215.4 psig
1190 psig with an acceptable range of 1154.3 psig to 1225.7 psig

As depicted above, if the SRVs setpoint were allowed to drift such that their actual setpoints approached the ±3% tolerance limit, sufficient setpoint overlap could exist such that the concept of banding would be lost. As stated in the licensee's submittal, when using the proposed acceptance criterion of ±3%, an individual SRV would not need to be recalibrated provided the as-found setpoint was found to be within the ±3% tolerance limit. However, if the as-found setpoint was found to be outside the ±3% tolerance limit, the SRV would be recalibrated to within ±1%.

The original design for the Perry SRVs assumed a limited number of SRV actuations for any given sequence. Given unchecked setpoint drift, the current request could mathematically result in the actuation of all SRVs for a single sequence as opposed to a discrete number of SRVs. The concern is whether simultaneous actuation of most or all of the SRVs would violate any of the original licensing basis. Specifically, the staff questioned whether the limiting structural loading analysis assumed that all 19 SRVs opened simultaneously. In this regard, the licensee has confirmed that the limiting structural loading calculations have assumed that all 19 SRVs open at the same time and in phase. USAR Figure 5.2-6B shows the reactor vessel pressure transient for the case of all MSIVs closing at full power and indicating that all 19 SRVs open at once.

In summary, the licensee has analyzed the structural loading assuming the worst case scenario in which all 19 SRVs opened simultaneously and determined that the resultant hydrodynamic loadings were within acceptable limits. Therefore, the staff finds this acceptable.

3.8. ECCS-LOCA

GE reviewed the LOCA analysis in the Perry USAR for the licensee to determine the effect of an increase in SRV opening pressures on ECCS performance. The limiting break LOCA, the DBA recirculation break, the small break LOCA, and the steam line break outside containment events were evaluated to determine the effects of the increased SRV setpoint tolerance. For the six SRVs equipped with Low - Low Set (LLS) logic, the increased SRV safety mode setpoint to +3% assumed for postulated small break LOCAs will only affect the timing of the first actuation. Once the logic is initiated, the opening and closing setpoints of these preselected SRVs are automatically reset to lower values by the LLS logic. This logic is not affected by the setpoint tolerance change since it acts on the relief mode of operation and not on the safety mode of operation. The acceptance criteria given in 10 CFR 50.46 are still satisfied for all break sizes and locations and hence the setpoint tolerance change for LOCA considerations is acceptable.

3.9. ATWS

The main steam isolation valve closure under ATWS conditions was reevaluated to support the tolerance increase of 3%. Using the staff-approved ODYN code and assuming two SRVs inoperable, the analysis shows that the vessel pressure reaches a maximum of 1344 psig, which is within the vessel overpressure criterion of 1500 psig for ATWS events. The long-term effect on suppression pool temperature due to 3% SRV tolerance is negligible because there is little change in the total energy discharged to the pool. The staff finds this acceptable.

3.10 TECHNICAL SPECIFICATION CHANGES

The setpoint tolerance in TS 3.4.4 is changed from ±1% to ±3%. This is acceptable as described in this SER.

The following note is added to TS Bases page B 3.4-21: " The safety lift setpoints will still be set within a tolerance of + or - 1 %, but the setpoints will be tested to within + or - 3 % to determine acceptance or failure of the as found valve lift setpoint." This change is acceptable to the staff as described in this SER.

By letter dated July 13, 1998, FirstEnergy submitted proposed changes to the Perry Technical Specifications. The proposed amendment will allow the licensee to increase the allowable SRV setpoint tolerance from ±1% to ±3%. In support of the modifications, the licensee has submitted plant specific analyses adequately addressing the six conditions identified in the Staff's SER for NEDC-31753P, " BWROG In-Service Pressure Relief Technical Licensing Topical Report." The proposed changes are, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in tne types, of any effluent 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (63 FR 43214). Accordingly, this amendment meets 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 this amendment.

6.0 CONCLUSION

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

Principal Contributors: George Thomas
Gary Hammer
Amira Gill

Date: March 3, 1999

ATTACHMENT A-1

Beaver Valley Power Station, Unit No. 1
Proposed License Amendment Request No. 256

The following is a list of the affected page:

Affected Page: 1-2

REPORTABLE EVENT

1.7 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each air lock is in compliance with the requirements of Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

(Proposed Wording)

INSERT 1

Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

ATTACHMENT A-2

Beaver Valley Power Station, Unit No. 2
Proposed License Amendment Request No. 126

The following is a list of the affected page:

Affected Page: 1-2

CONTAINMENT INTEGRITY (Continued)

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each air lock is in compliance with the requirements of Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows, O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. **Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.** The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY, including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

INSERT 1

Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

ATTACHMENT B

Beaver Valley Power Station, Unit Nos. 1 and 2
Proposed License Amendment Request Nos. 256 and 126
REVISION OF DEFINITION OF CHANNEL CALIBRATION

A. DESCRIPTION OF AMENDMENT REQUEST

The proposed amendment would revise the existing definition of Channel Calibration in Technical Specification 1.9 to add the following two sentences. §Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required Channel Calibration shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.§

B. DESIGN BASES

Resistance Temperature Detectors [RTDs] and thermocouples are used in various circuits at Beaver Valley. These sensors are used in the Overtemperature ΔT, Overpower ΔT, the Engineered Safety Features [ESF] P-12 interlock, the Meteorological Tower ΔT, remote shutdown panel, Reactor Coolant System [RCS] hot and cold leg temperatures, RHR heat exchanger outlet temperature, RCS subcooling margin indication, in-core temperature indication, Reactor Vessel Level Indication System temperature compensation, and hydrogen recombiner heater outlet temperature instrumentation systems. The surveillances of these instrumentation systems are specified in Technical Specification Tables 4.3-1, 2, 5, 6, & 7. These tables require Channel Calibration for each of these instruments.

C. JUSTIFICATION

The present definition of Channel Calibration requires that Channel Calibration shall encompass the entire channel, including the sensor, and shall include the Channel Functional Test. RTDs and thermocouples are not adjustable during calibration and in the case of in-core thermocouples are not accessible without significant exposure to ionizing radiation. The Standard Technical Specifications, as described in NUREG 1431 provides a definition of Channel Calibration that permits the use of a qualitative assessment of RTD or thermocouple behavior in lieu of normal calibrations. The inplace cross calibration of RTDs and thermocouples is an acceptable method of performing a qualitative assessment of these sensors' behavior as delineated in NRC Branch Technical Position HICB-13, §Guidance on Cross Calibration of Protection System Resistance Temperature Detectors,§ Rev. 4, June 1997. Acceptance of cross calibration of RTDs and thermocouples by averaging all sensors measuring the same variable and comparing each individual sensor with the average has been approved by the NRC in the past. This has occurred in the case of license amendments which have adopted~~requests for adoption of~~ the Standard Technical Specification [STS] definition of Channel Calibration for Units 1 and 2 at Salem (Amendments 191 and 174), Susquehanna (Amendments 133 and 102) and La Salle (Amendments 120 and 105), as recently as September 1997.

D. SAFETY ANALYSIS

This proposed revision to the Beaver Valley Unit 1 and 2 Technical Specifications is being requested to better account for standard industry methodology for temperature sensor Channel Calibration. This methodology avoids unnecessary removal or replacement of these sensors from their installed location for calibration. Removal and installation of RTDs or thermocouples solely for the purpose of calibration could introduce errors, cause sensor damage, and increase personnel exposure to ionizing radiation. Most of these sensors are located in systems containing radioactive fluid.

In order to confirm the calibration of instrument channels having RTDs or thermocouple temperature sensors Duquesne Light Company [DLC] will be performing inplace qualitative assessments of sensor behavior. The RCS loop RTDs are cross calibrated in accordance with current procedures. The other required Technical Specification RTD and thermocouple qualitative assessments will be completed prior to changing modes from the present plant operating Mode, which is Mode 5, cold shutdown. Subsequent RTD and thermocouple qualitative assessments will be performed consistent with the frequency specified in the current Technical Specifications. If calibration of a temperature sensor is confirmed by inplace qualitative assessment using cross calibration, and if that temperature sensor must be replaced, the next required Channel Calibration will include an inplace cross calibration which compares the similarly located sensing elements with the recently installed sensing element.

The issue of cross calibration was addressed in NUREG/CR-5560, ▒Aging of Nuclear Plant Resistance Temperature Detectors,▒ which recognizes that on-line cross calibration can be a reasonable method for temperature detector calibration. However, as stated in NUREG/CR-5560, to perform in-situ calibration would normally require one or more newly calibrated sensors to be used as a reference. Without a reference, the cross calibration might not account for common cause drift. The cross calibration technique assumes that the average of the sensor measurements represents the true process temperature and that sensor drift is random and not systematic. The results of studies referenced in NUREG/CR-5560 indicated that sensor drift is random in nature. Therefore, the cross calibration technique is an acceptable method of performing the qualitative assessment of temperature sensor behavior discussed in the attached proposed change to the

Beaver Valley Unit 1 and 2 Technical Specification, Section 1.9, Channel Calibration definition.

In the NRC letter from Donald Brinkman to DLC [J. E. Cross] of June 11, 1998, the NRC stated that "...Adoption of improved STS surveillance requirements (SR) for RTDs and thermocouples includes additional testing requirements specified in the SR notes such as verifying rate lag compensation for flow from the core to the RTDs." In a telephone call between a member of the DLC staff and a member of the NRC Technical Specification Branch on June 16, 1998, it was stated that this provision of the NRC June 11 letter referred to the STS Bases, assuming the RTDs are in an RCS bypass loop. The STS Bases for SR 3.3.1.12 states, "This test will verify the rate lag compensation for flow from the core to the RTDs." In view of the fact that neither unit at the Beaver Valley Station has a bypass loop for the RCS RTDs, they are in thermal wells directly immersed in the main RCS loop, no additional response to the above quoted statement in the June 11 NRC letter is being provided.

The modified Channel Calibration definition is functionally consistent with the Improved Standard Technical Specifications [ISTS], provides clarification of the Channel Calibration requirements and maintains compliance with the applicable specification operability requirements. This change does not affect the system description or UFSAR accident analyses; therefore, this change has been determined to be safe and will not reduce the safety of the plant.

E. NO SIGNIFICANT HAZARDS EVALUATION

The no significant hazard considerations involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

> The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:
>
> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the no significant hazards consideration standards.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change is administrative in nature. It does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident nor alter the conditions or assumptions in any of the Updated Final Safety Analysis Report [UFSAR] accident analyses. The revised definition would eliminate unnecessary and potentially damaging removal of resistance temperature detector (RTD) or thermocouple sensors in order to perform calibrations that are not technically possible. Therefore, it can be concluded that the proposed changes do not involve any increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No new failure modes have been defined for any plant system or component important to safety nor has any new limiting failure been identified as a result of the proposed changes. There will be no change in the requirement to assess the entire RTD or thermocouple channel behavior including the sensor, alarm, display, and/or trip function. Therefore, it can be concluded that the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

   The proposed change is administrative in nature. Assessment of channel behavior, including sensors, will continue to be required. The addition to the Channel Calibration definition will provide greater flexibility in the use of the provision for surveillance testing, and will have no adverse effect on safety. Also, the inplace qualitative assessment obviates the need to remove the RTDs or thermocouples from their installed location, thereby eliminating the possibility of damaging them during removal. Therefore, it can be concluded that the proposed changes do not involve any reduction in a margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the considerations expressed above, it is concluded that the activities associated with this license amendment request satisfy the requirements of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL CONSIDERATION

This license amendment request changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. It has been determined that this license amendment request involves 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. This license amendment request may change requirements with respect to installation or use of a facility component located within the restricted area or change an inspection or surveillance requirement; however, the category of this licensing action does not individually or cumulatively have a significant effect on the human environment. Accordingly, this license amendment request meets 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 this license amendment request.

H. UFSAR CHANGES

No UFSAR changes are required.

ATTACHMENT C-1

Beaver Valley Power Station, Unit No. 1
Proposed License Amendment Request No. 256

Applicable Typed Pages

ATTACHMENT TO LICENSE AMENDMENT NO. 216

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of Appendix A, Technical Specifications, with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
I	I
1-2	1-2
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REPORTABLE EVENT

1.7 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each air lock is in compliance with the requirements of Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

LEAKAGE

1.14 LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be Pressure Boundary LEAKAGE, or

3. Reactor Coolant System LEAKAGE through a steam generator to the secondary system.

b. Unidentified LEAKAGE

Unidentified LEAKAGE shall be all LEAKAGE (except reactor coolant pump seal water injection or leakoff) that is not Identified LEAKAGE.

c. Pressure Boundary LEAKAGE

Pressure Boundary LEAKAGE shall be LEAKAGE (except steam generator tube LEAKAGE) through a nonisolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

1.15 THROUGH 1.17 (DELETED)

QUADRANT POWER TILT RATIO (QPTR)

1.18 QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The DOSE EQUIVALENT I-131 is calculated with the following equation:

$$C_{I-131_{D.E.}} = C_{I-131} + \frac{C_{I-132}}{170} + \frac{C_{I-133}}{6} + \frac{C_{I-134}}{1000} + \frac{C_{I-135}}{34}$$

Where C is the concentration, in microcuries/gram of the iodine isotopes. This equation is based on dose conversion factors derived from ICRP-30.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals;

b. The testing of one (1) system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

REACTOR TRIP SYSTEM RESPONSE TIME

1.22 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.23 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

AXIAL FLUX DIFFERENCE

1.24 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

PHYSICS TESTS

1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.26 $\bar{E}$ shall be the average sum (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ATTACHMENT C-2

Beaver Valley Power Station, Unit No. 2
Proposed License Amendment Request No. 126

Applicable Typed Pages

ATTACHMENT TO LICENSE AMENDMENT NO. 93

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

Replace the following pages of Appendix A, Technical Specifications, with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
I	I
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Summary of Feedback from the NRC/FENOC Licensing Workshop

The following discussion is a summary of the written feedback forms that were distributed to workshop participants.

Participants were asked to rate the workshop on a scale of 1 to 10 on areas such as accomplishment of objective, and completeness and suitability of subject matter. The average grade given was an 8. When asked for an overall rating for the workshop, the average grade was also an 8.

The written comments are summarized below:

- Most participants appreciated the open and honest dialog with the NRC and the chance to interact with their NRR counterparts.
- The discussions on current regulatory issues and initiatives was useful and several participants suggested holding meetings such as this every 1 or 2 years to keep informed of new developments.
- Most participants found the breakout session in which a recent licensing submittal was discussed to be one of the most useful aspects of the workshop since this affected their daily jobs most.
- The fact that the discussions were at the working level made for more open and useful dialog.
- Several participants observed that project managers rarely travel to the sites anymore and that it was beneficial to have their staffs meet the project manager in person.
- One participant suggested that an example of a good NOED submittal would have been helpful to supplement the NOED reference booklet that was handed out.
- One participant suggested that a reviewer from NRR's technical staff provide their observations and needs regarding licensing submittals.

Some of the questions/concerns that were expressed to the NRC staff during the workshop included:

- Recent statements by the staff regarding the use of Oath and Affirmation have caused confusion within the industry. The NEI representative stated that this issue is being addressed.
- Extensive discussion focused on the staff's process of upgrading the improved standard technical specifications. Specifically, there have been recent examples of the staff approving technical specification task force (TSTF) "travelers" (i.e., upgrades) for some licensees but then changing their position and not approving the same traveler for other licensees.
- Several participants expressed their frustration in dealing with ADAMS.

ATTENDANCE LIST

FENOC

James Emley
David Lockwood
Gregory Dunn
Kenneth Russell
Tom Cosgrove
Tony Dometrovich
Steve Sarver
Kevin Spencer
Mark Riemer
Mike Leisure
Frank Kennedy
Brian Sepekek
Ben Spiesman
Bradley Ferrell
Dale Wuokko
Gary Rhoads
Kenneth Meade
Jane Mallemee

NRC

Anthony Mendiola
Daniel Collins
Stephen Sands
Douglas Pickett

Nuclear Energy Insitute

Mike Schoppman

State of Pennsylvania

Larry Ryan

CONTAINMENT INTEGRITY (Continued)

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each air lock is in compliance with the requirements of Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

LEAKAGE

1.14 LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be Pressure Boundary LEAKAGE, or

3. Reactor Coolant System LEAKAGE through a steam generator to the secondary system.

b. Unidentified LEAKAGE

Unidentified LEAKAGE shall be all LEAKAGE (except reactor coolant pump seal water injection or leakoff) that is not Identified LEAKAGE.

c. Pressure Boundary LEAKAGE

Pressure Boundary LEAKAGE shall be LEAKAGE (except steam generator tube LEAKAGE) through a nonisolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

1.15 THROUGH 1.17 (DELETED)

QUADRANT POWER TILT RATIO (QPTR)

1.18 QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Regulatory Guide 1.109, 1977 or TID 14844.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals;

b. The testing of one (1) system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

REACTOR TRIP SYSTEM RESPONSE TIME

1.22 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.23 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

AXIAL FLUX DIFFERENCE

1.24 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two-section excore neutron detector.

PHYSICS TESTS

1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.26 $\bar{E}$ shall be the average sum (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

SOURCE CHECK

1.27 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

PROCESS CONTROL PROGRAM

1.28 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

1.29 DELETED

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive

OFFSITE DOSE CALCULATION MANUAL (ODCM) (Continued)

Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.6 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.10 and 6.9.1.11.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting Primary Coolant System offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE-PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions, in such a manner that replacement air or gas is required to purify the confinement.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

MAJOR CHANGES

1.35 MAJOR CHANGES to radioactive waste systems (liquid, gaseous and solid), as addressed in the PROCESS CONTROL PROGRAM, shall include the following:

MAJOR CHANGES (Continued)

1) MAJOR CHANGES in process equipment, components, structures, and effluent monitoring instrumentation from those described in the Final Safety Analysis Report (FSAR) or the Hazards Summary Report and evaluated in the staff's Safety Evaluation Report (SER) (e.g., deletion of evaporators and installation of demineralizers; use of fluidized bed calciner/incineration in place of cement solidification systems);

2) MAJOR CHANGES in the design of radwaste treatment systems (liquid, gaseous, and solid) that could significantly increase the quantities or activity of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3) Changes in system design which may invalidate the accident analysis as described in the SER (e.g., changes in tank capacity that would alter the curies released); and

4) Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of temporary equipment without adequate shielding provisions).

MEMBER(S) OF THE PUBLIC

1.36 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries and persons who traverse portions of the site as the consequence of a public highway, railway, or waterway located within the confines of the site boundary. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

CORE OPERATING LIMITS REPORT

1.37 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.12. Plant operation within these operating limits is addressed in individual specifications.

ATTACHMENT D-1

Beaver Valley Power Station, Unit No. 1
Proposed License Amendment Request No.

[redacted]

Applicable UFSAR Changes

ATTACHMENT D-2

Beaver Valley Power Station, Unit No. 2
Proposed License Amendment Request No.

Applicable UFSAR Changes

LATF ACTIVITIES

EMERGING NRC PROCESSES PART 2

Doug Pickett, Senior Project Manager
Project Directorate 3
Division of Licensing Project Management

DISCUSSION ITEMS

- New Oversight Process
- ADAMS
- Electronic Information Exchange

NRC REACTOR OVERSIGHT PROCESS

- New process uses more objective, timely, and safety-significant criteria in assessing plant performance

- New program tested at 13 reactors at 9 sites in 1999

- All reactors fall within new oversight process on April 2, 2000

KEY FEATURES OF THE PROGRAM

- Inspections focus on activities where potential risks are greater
- Greater attention applied on facilities with performance problems while normal level of attention applied to good performers
- Uses objective measurements of plant performance

KEY FEATURES OF THE PROGRAM (continued)

- Provides timely and understandable assessments of plant performance to both the public and the industry
- Reduces unnecessary regulatory burden
- Responds to violations of regulations in a predictable and consistent manner that reflects potential safety impact of violations

MONITORS PERFORMANCE IN THREE BROAD AREAS

- Reactor Safety - Avoiding accidents and reducing the consequences of accidents

- Radiation Safety - For both plant workers and the public during routine operation

- Protection of the Plant Against Sabotage or Other Security Threats

FOCUS ON SEVEN SPECIFIC "CORNERSTONES"

- Initiating Events - Operations or events that could lead to an accident
- Mitigating Systems - Measures the function of safety systems
- Barrier Integrity - Fuel rods, reactor vessel, and containment
- Emergency Preparedness - Effectiveness of emergency plan

FOCUS ON SEVEN SPECIFIC "CORNERSTONES" (cont)

- Occupational Radiation Safety - Monitors effectiveness of plant program to control and minimize worker exposures
- Public Radiation Safety - Minimize offsite releases
- Physical Protection - Physical Security/Fitness for Duty

"CROSS-CUTTING" ELEMENTS

- These affect each cornerstone
- Human Performance
- Management Attention to Safety and Workers Ability to Raise Safety Issues - The "Safety-Conscious Work Environment"
- Finding and Fixing Problems - Correction Action Programs

PERFORMANCE INDICATORS (PIs)

- Uses objective data to monitor performance within each cornerstone
- Data to develop PIs generated by licensees and provided on a quarterly basis
- Each PI measured against established threshold which relate to safety
- PIs to be evaluated and integrated with NRC Inspection Findings

COLOR CODING SYSTEM

- Green - Performance within expected performance level. Related cornerstone objectives met.

- White - Performance outside expected performance level. Related cornerstone objectives met

- Yellow - Performance with minimal reduction in safety margin. Related cornerstone objectives met

- Red - Performance with significant reduction in

SCOPE OF INSPECTIONS

- Continues to rely on regional and resident inspectors
- Baseline inspections common to all plants
- Baseline inspections consist of:
  - Areas not covered by PIs
  - Verifies licensee PI data
  - Reviews licensee's effectiveness in finding and resolving problems

SCOPE OF INSPECTIONS (cont)

- Baseline inspections based on cornerstone areas and focus on activities and systems that are risk significant
- Uses risk-informed approach to select areas to inspect within each cornerstone

SIGNIFICANT DETERMINATION PROCESS

- SDP helps inspectors determine safety significance of inspection findings
- Used as a screening tool - "Green" findings not pursued
- Findings having an effect on plant risk will be subject to a more thorough risk assessment

ASSESSING PLANT PERFORMANCE

- Quarterly review of plant performance via PIs and inspection findings
- Every 6 months performance review will include planning of inspections for the next 12 months
- Final quarterly plant performance review of the year will included detailed assessment over previous 12 months
- Annual plant performance reports listed on NRC web site/ Public Meeting
- NRC senior management review of plants with significant performance problems

NRC Response Plan or "Action Matrix"

Assessment of Plant Performance (in order of increasing safety significance)	NRC Response
I. All performance indicators and cornerstone inspection findings GREEN !Cornerstone objectives fully met.	! Routine inspector and staff interaction ! Baseline inspection program ! Annual assessment public meeting
II. No more than two WHITE inputs in different cornerstones !Cornerstone objectives fully met.	**Response at Regional level** !Staff to hold public meeting with utility management !Utility corrective action to address WHITE inputs !NRC inspection followup on WHITE inputs and corrective action
III. One degraded cornerstone (two WHITE inputs or one YELLOW input or three WHITE inputs in any strategic area) !Cornerstone objectives met with minimal reduction in safety margin	**Response at Regional level** !Senior regional management to hold public meeting with utility management !Utility to conduct self- assessment with NRC oversight !Additional inspections focused on cause of degraded performance
IV. Repetitive degraded cornerstone, multiple degraded cornerstones, or multiple YELLOW inputs, or one RED input !Cornerstone objectives met with longstanding issues or significant reduction in safety margin	**Response at Agency level** !Executive Director for Operations to hold public meeting with senior utility management !Utility develops performance improvement plan with NRC oversight !NRC team inspection focused on cause of degraded performance !Demand for Information, Confirmatory Action Letter, or Order
V. Unacceptable Performance !Unacceptable reduction in safety margin	**Response at Agency level** !Plant not permitted to operate !Commission meeting with senior utility management !Order to modify, suspend, or revoke license

AGENCYWIDE DOCUMENTS

ACCESS & MANAGEMENT SYSTEM

ADAMS

DEFINITION OF ADAMS

The policies, processes, and software tools to manage unclassified, official program, and administrative records of lasting business value to the NRC in an electronic rather than paper-based environment

IMPORTANCE OF ADAMS

- The NRC will achieve productivity gains
- Improve communication within the NRC and with licensees and other stakeholders
- Make public documents available to the public via the Internet
- Submittals to the NRC can be in electronic form via the internet

WHAT WILL ADAMS CHANGE

- Voluntary electronic submission of documents from the NRC stakeholders
- Electronic distribution of documents
- The electronic image of the document will be the official agency record
- Electronically route, assign, concur in documents, and track status
- Retrieve full text and images of documents from electronic repository

BENEFITS OF ADAMS

- Improved integrity of information
- Faster, broader access to documents
- Streamlined concurrence; Improved tracking
- Security/access control
- Eventual elimination of paper copy
- Documents available much faster
- Reduced information management costs

IMPLEMENTATION STRATEGY

There will be a phased deployment of users and system capabilities that has already begun

USE OF ELECTRONIC MEDIA

- Provide NRC with Electronic Copy of License Submittals
- Information made available to the NRC quicker
- Preparation of Notices, Safety Evaluations,
- Amendments easier
- Information posted on ADAMS for easier access
- NRC working on Policies for Electronic Information Exchange - Voluntary Participation

Electronic Information Exchange (EIE)

- Must register to become Electronic Trading Partner
- NRC is reviewing the surety levels required for submitted documents to establish the requirements for handling them in electronic form.
- Rulemaking will be Initiated to Allow Electronic Filing (expected July 2000)
- NRC will be responsible for distribution
- Externally generated documents will be distributed using ADAMS software.

Electronic Information Exchange (EIE) (continued)

- Distribution outside the NRC, either electronic or paper form depending on the recipient
- Very large documents would be submitted via the U.S. mail on CD-ROM (larger than 2 MB)
- Smaller documents, the majority, would be submitted electronically via NRC's EIE program at our web site
- NRC's current plan is to accept documents in PDF, MS Word, and Word Perfect formats