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

August 18, 1989

Docket No. 50-267

MEMORANDUM FOR: Seymour H. Weiss, Director
Non-Power Reactor, Decommissioning
and Environmental Project Directorate
Division of Reactor Projects - III,
IV, V and Special Projects

FROM: Kenneth L. Heitner, Project Manager
Non-Power Reactor, Decommissioning
and Environmental Project Directorate
Division of Reactor Projects - III,
IV, V and Special Projects

SUBJECT: SUMMARY OF MEETING WITH PUBLIC SERVICE COMPANY OF COLORADO (PSC) TO DISCUSS DEFUELING OF FORT ST. VRAIN (FSV) JULY 18, 1989 (TAC NO. 73124)

This meeting was requested by PSC to further discuss issues related to the defueling of FSV. The attendees at this meeting are listed in Enclosure 1. Material supplied by PSC at this meeting is Enclosure 2.

Background

This meeting was a followup to letters dated May 11 and May 15, 1989 submitted to the staff by PSC. In these letters PSC indicated two fundamental changes in its approach to the defueling of FSV and its conduct of operations during the defueling period. The first change is that PSC would conduct the defueling of FSV under 10 CFR 50.59. The second is that PSC would not request further Technical Specification changes specifically for defueling, or as part of the Technical Specification Upgrade Program (TSUP). These viewpoints were presented by PSC at a meeting with NRC managers on May 11, 1989.

PSC's Restatement of Position

PSC restated their position that NRC approval was not required for defueling of Fort St. Vrain. Key points in PSC's presentation were that:

- There were no unreviewed safety questions,
- There would be an increased safety margin during defueling, and
- That defueling safety concerns were adequately controlled by existing Technical Specification requirements.

In their discussion of the proposed defueling, PSC noted the importance of maintaining the core configuration to be a right circular cylinder. This

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geometry was capable of being analyzed by accepted methods. PSC also noted their interpretation of Interim Limiting Condition for Operation (LCO) 3.1.6 as being only applicable to the portion of the core containing fuel. The control rods would be withdrawn in regions where dummy fuel elements were present. PSC admitted the proposed dummy fuel elements were not discussed (or described) in Section 6.0 of the FSV Technical Specifications (TS). However, PSC stated that the safety basis were the shutdown margin requirements in other portions of the TS. PSC also noted that some uncertainty still existed about the ability of the excore startup neutron detectors to monitor the defueling process. Potentially, the count rate for detectors could fall below TS allowable values before the core was subcritical with all rods out. PSC was considering whether a license amendment would be needed later in the defueling process. PSC also stated that they had not decided whether to reevaluate control rod worths prior to starting defueling.

Staff's Statement of Concerns

The staff presented its concerns with respect to the proposed method of defueling. These concerns would have to be adequately addressed in a licensee 50.59 evaluation. The staff had considered a number of other documents, including PSC letters dated January 20 and June 16, 1989 and the summary of the previous meeting with PSC on defueling dated March 13, 1989 in identifying these issues.

First, the proposals for defueling the reactor are significantly different from the original reactor fueling as defined in the FSV Final Safety Analysis Report (FSAR) (Section 13.3). The fueling was done by layers with temporary absorber strings. The defueling is proposed to be by core regions with a radially inward pattern. Hence, the defueling would differ significantly from previously described fueling and refueling activities.

The second issue of concern was the ability of the startup detectors to provide adequate measurement and monitoring of the core's subcritical configuration during the defueling process. Again, the proposed use of the startup detectors is different from the approach used during reactor fueling where detectors were placed in the reactor core. The changes in core geometry during defueling are significant compared to those involved with a normal refueling. The capability of the startup channels to detect criticality would be affected by the reduced production of neutrons per source neutron. The adequacy of the source range trip, currently set at 10^5 counts per second (cps), could also be significantly affected. These questions are unique to the defueling scenario. The purpose of the startup detectors and the associated trip setpoint is to scram the reactor should an inadvertent criticality occur during the defueling operation. (PSC has also noted the problem of maintaining an adequate count rate in the detectors.)

The third issue relates to the description of the Reactor Core - Design Features. Section 6.0 of the Fort St. Vrain FSAR addresses design features. In particular, Section 6.1 addresses the design features of the reactor core. The objective of this section states: "to define vital design characteristics of the reactor core to control changes in the design features." In the

discussions on March 7, 1989 concerning FSV defueling, PSC proposed the use of dummy fuel blocks containing boron material. As the defueling process proceeded, the absence of fuel and its replacement with the dummy fuel blocks would play an increasing and significant role in maintaining the core subcritical (i.e., providing reactivity control). Once a core region is defueled, the dummy fuel blocks provide the negative reactivity. In addition, the dummy fuel blocks maintain the structural integrity of the core. However, the dummy fuel blocks are not described in TS Section 6.1 - Reactor Core - Design Features. The materials of construction are changed from those specified.

Finally, by letter dated July 10, 1985, PSC committed to operating FSV under the Interim Technical Specifications for Reactivity Control (attached to that letter). A literal reading of Certain Interim TS are potentially inconsistent with PSC's proposed defueling approach. For example, Interim TS 3.1.6.A.1 only allows up to two control rod pairs to be withdrawn for refueling. PSC's proposed defueling would require more than two control rod pairs withdrawn. The 50.59 evaluation associated with defueling would have to clarify the intent of the interim TS for this proposed activity and to show that the commitment to operate under the interim TS is not changed in a manner that would require NRC review and approval.

Conclusions

No conclusions were reached relative to staff agreement with PSC defueling FSV under the terms of 10 CFR 50.59. The staff stated that the decision to proceed with evaluating the defueling under 10 CFR 50.59 remained with PSC. However, PSC in their 10 CFR 50.59 analysis will have to address the issues the staff identified including how the requirements of 10 CFR 50.59 have been met. The staff requested that PSC provide a copy of the defueling Safety Analysis Report (SAR) when it was complete in any event. PSC agreed to this request. PSC also agreed to provide the SAR for plant coastdown past 300 equivalent full power days.

Technical Specification changes under the Technical Specification Upgrade Program were not discussed at this meeting.

/s/

Kenneth L. Heitner, Project Manager
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Enclosures:
As stated
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B. Grimes	ACRS (10)	L. Kopp	G. Holahan
T. Westerman, RIV	J. Partlow	E. Tomlinson	T. Martin (Region IV)

DOCUMENT NAME: FSV 8980
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PD4/LA*	PD4/PM*	PD4/D*	(A)ADR4
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Enclosure 1

Attendees at NRC-PSC Meeting of July 18, 1989

Name	Organization
Ken L. Heitner	NRC/NRR/PD-IV
Don Warenbourg	PSC
A. Clegg Crawford	PSC
Charles H. Fuller	PSC
H. L. Brey	PSC
R. H. Vollmer	PSC/Tenera
D. Alberstein	General Atomics
Danialle Weaver	Nucleomes Week
Larry Kopp	NRC/NRR/SRXB
Gary Holahan	NRC/NRR/DRSP
Tom Westerman	NRC/RIV
James Partlow	NRC/NRR
Ed Tomlinson	NRC/NRR/PD-IV

Enclosure 2

AGENDA

NRC-PSC EXECUTIVE MEETING
JULY 18, 1989

I. INTRODUCTION/OBJECTIVE	Crawford
II. DEFUELING STATUS	Warembourg
III. DEFUELING SAR SUMMARY	Fuller
IV. CONCLUSIONS	Crawford
V. PROPOSED ACTIONS	Crawford
VI. DISCUSSION	

INTRODUCTION

MEETING OBJECTIVE

- CLARIFY THE NEED FOR NRC APPROVALS, IF ANY, TO DEFUEL FORT ST. VRAIN

INTRODUCTION (CONTINUED)

PROPOSED ACTIONS

- SUBMIT FINAL DEFUELING SAR FOR NRC INFORMATION
  - Will be submitted with a cover letter including:
    - Brief description of defueling process
    - Assumptions and basis for 50.59 conclusions
- SET UP A JOINT WORKING MEETING TO RESOLVE REMAINING TECHNICAL ISSUES

PSC POSITION

- NRC APPROVAL NOT REQUIRED FOR DEFUELING
  - Not required by regulations
  - Not required for typical LWR defueling
  - No defueling unreviewed safety questions
  - Increased safety margin during defueling
    - Less reactive
    - Fission products decaying
    - Reduced heat generation
  - No immediate threat to public health or safety
  - Defueling safety concerns adequately controlled by existing Technical Specification requirements

DEFUELING OVERVIEW

- Defueling milestones
  - Be ready to begin defueling as early as November 1, 1989
  - Base case defueling plans begin defueling January 2, 1990
  - Optimum case begin defueling July 1, 1990
- Strategy
  - Keep defueling similar to refueling
  - Defuel by region
  - Defuel by ring outer to inner to maintain validity of computer models
  - Replace fuel elements with boronated, HLM graphite defuel elements
  - Utilize existing Technical Specifications
- SAR for coast down
- SAR for defueling

SAFETY ANALYSIS REPORT

FOR REACTOR DEFUELING

CONTENTS

DEFUELING SEQUENCE OBJECTIVE

- Utilize a shrinking core concept to ensure a core geometry consistent with established Fort St. Vrain core physics analysis models

- Ensure sufficient shutdown margin at all points in the sequence

- Ensure a neutron count rate on the startup channels that is adequate for monitoring core reactivity until such monitoring is no longer needed

- Minimize the number of fuel handling machine movements

- Provide for efficient fuel deck logistics

CORE PREPARATIONS

- Remove metal clad (boronated) top reflector Regions 3, 6, 10, and 16

- Install neutron sources in Regions 3 and 6

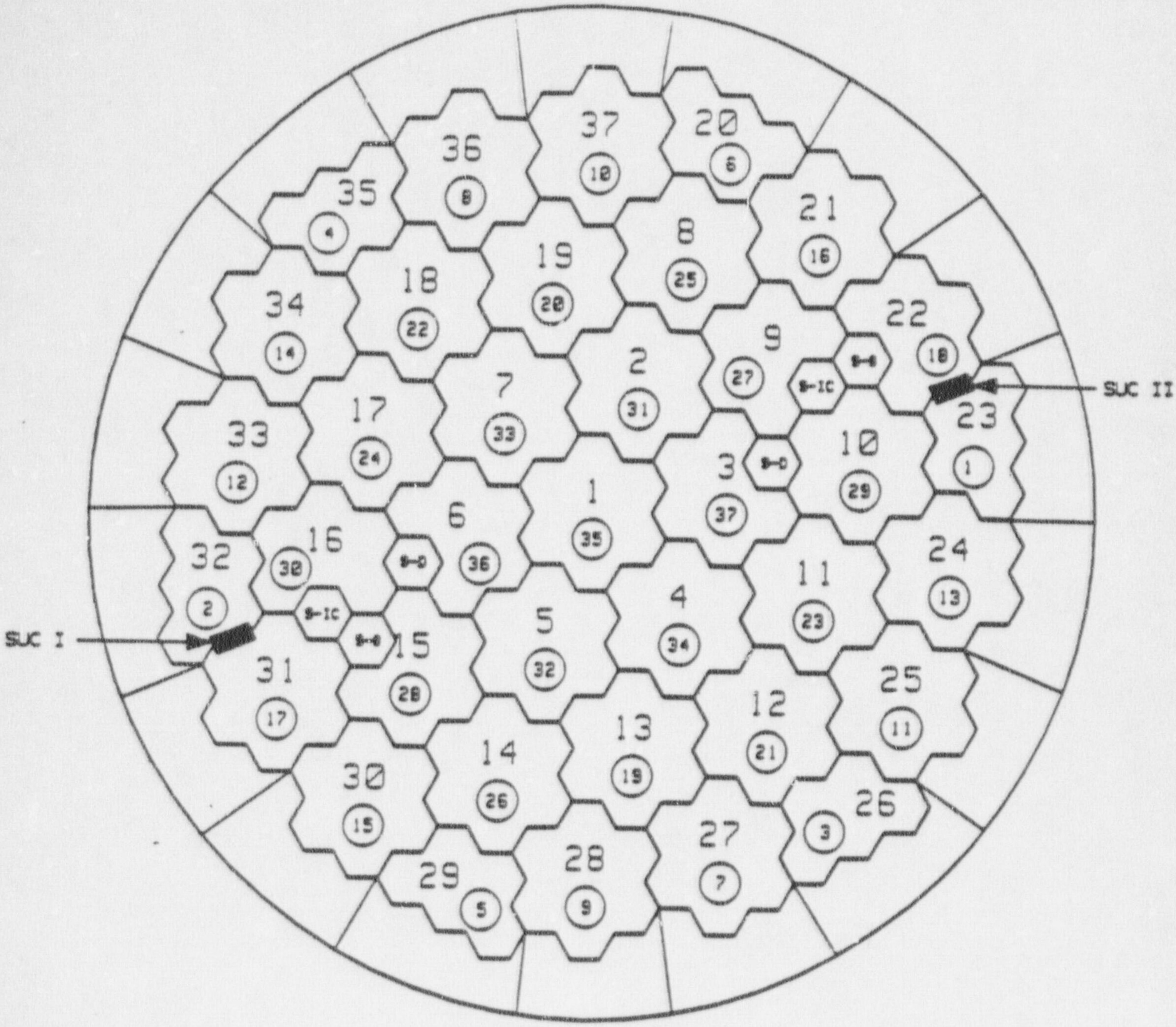
- Replace metal clad top reflector in Regions 3, 6, 10, and 16 with non-boronated elements

DEFUELING SEQUENCE

- Defuel by ring - outer to inner

- Shrinking core concept maintaining right circular cylinder configuration

- Regions 3 and 6 which contain neutron sources are the last regions to be defueled



LEGEND

REGION NUMBER 30

SEQUENCE NUMBER (37)

(9-IC) SOURCE ELEMENT

IC INITIAL CORE

8,9 SEGMENT 8 OR 9

D DEFUELING

Figure 2-1 Reactor Core Defueling Sequence

DEFUELING ELEMENTS

- HLM equivalent graphite elements
  - Satisfies all reactor physics, thermal and overall environmental requirements
  - Similar in structural strength to H-327 or H-451 and conservatively meets or exceeds all thermal hydraulic requirements
  - HLM graphite is currently used for permanent side reflector elements
  - Boron is presently installed in the side reflectors
- Boron carbide lumped poison pins
  - Equivalent of 100 ppm is adequate for reactivity control during defueling
  - Poison loading provides for the equivalent of 350 ppm for conservatism

- A region filled with defueling elements is at least equivalent to the control rod worth

- Overall design equivalent to existing fuel elements with the exception of inner ring of coolant holes and use of blind holes for lumped poison pins

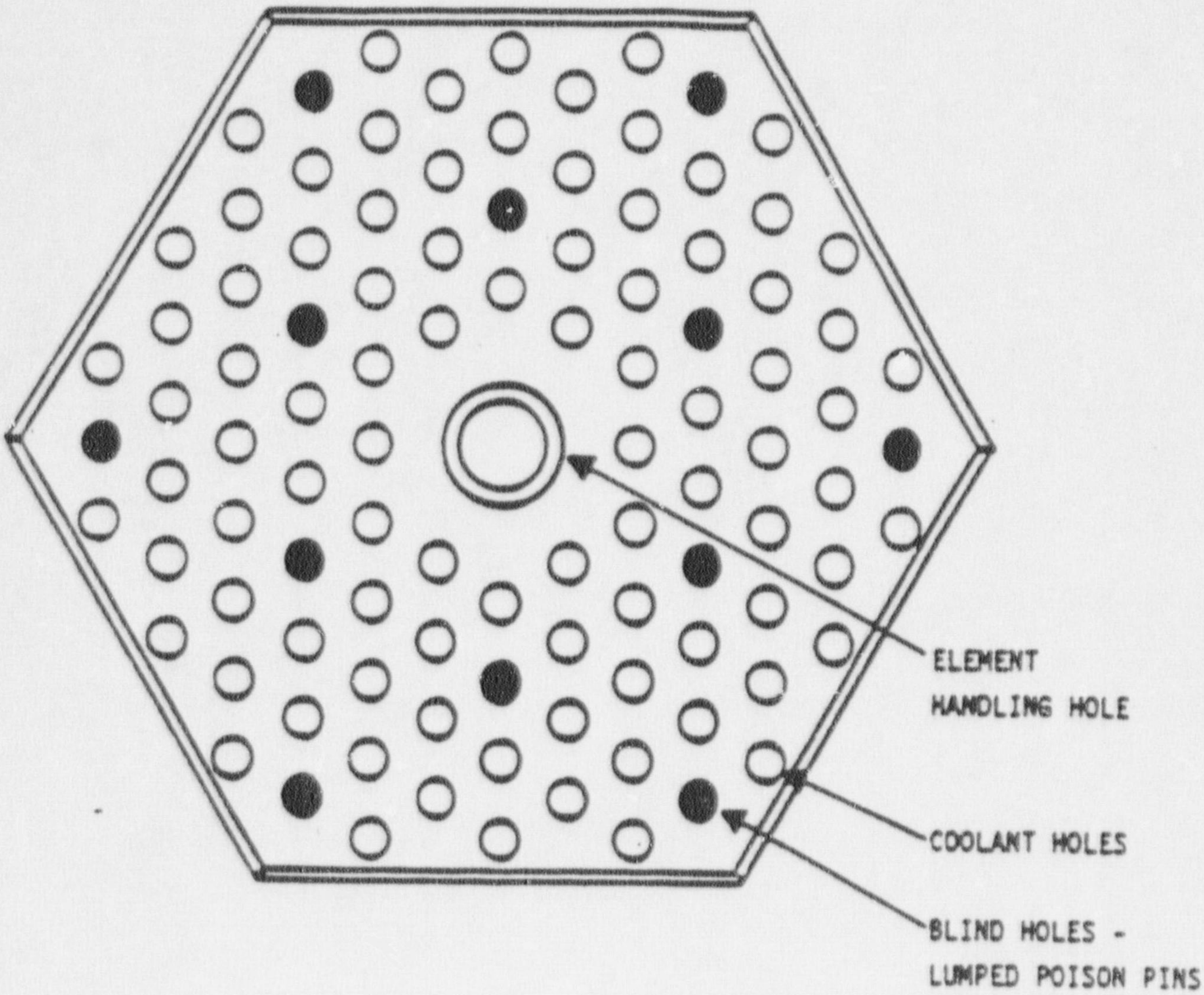


Figure 2-2 Defueling Element Top View (Dowels Not Shown)

REACTIVITY MONITORING

- Accomplished using the existing startup channels

- Count rate will be further enhanced by inserting additional neutron sources in Regions 6 and 3

- Before proceeding with the defueling of a region, a shutdown margin confirmation test is done in accordance with the current Technical Specifications. Count rates are monitored during this test

- Count rates are recorded before and after a region is defueled

- The current monitoring requirement and shutdown margin assessment testing is relaxed when:

  - The calculated Keff of the remaining fuel is less than or equal to 0.95 with all rods withdrawn

  - Physical demonstration of subcriticality is performed by withdrawing all control rods and verifying subcriticality. This would be the final shutdown margin verification test

SHUTDOWN MARGIN ASSUMPTIONS

- Uses the before-mentioned sequence

- Analysis was performed at 155 EFPD (core burnup is now over 200 EFPD)

  - Approximately 0.5% per 50 EFPD burnup

- Two rods in the sequence are withdrawn. Calculations also assumed that the Region 1 Rod is withdrawn. Current intention is to not withdraw the Region 1 Rod.

- Use 12 LPP/block design using blind coolant holes.

  - Minimum diameter
  - Minimum stack height
  - Minimum concentration

- Calculations performed with gauge code model

FINAL SHUTDOWN MARGIN VERIFICATION

- The criterion is calculated Keff less than or equal to 0.95 with all rods out

- With increasing core burnups, the criterion is met with more regions of fuel remaining

- Shutdown is after steady state operation at 82% power

CONCLUSION:

At 200 EFPD, the criteria is met with 8 regions of fuel left (Regions 1 through 7, and 16)

ACCIDENT EVALUATION

- Accidents involving depressurization are not credible since the PCRV will be maintained near atmospheric

- LOFC with one PCRV liner cooling loop was analyzed. The cooling loop was started within 24 hours of the LOFC which occurred 100 days after shutdown

  RESULTS:

  - All core internal temperatures are lower than the temperatures experienced during normal operation

Seismic event was analyzed with a region of fuel removed

RESULTS:

- A fallen element may break, fuel particle integrity remains intact, and fuel temperatures stay below 2,900 degrees

- Core support block would be damaged only if struck in the center. However, core support posts are not damaged and the overall core support structural integrity is maintained

- No chance for recriticality

- No safety consequences but PSC must deal with a different cleanup problem

ACCIDENT EVALUATION (CONTINUED)

- Inadvertent criticality accident. A postulation was made that during the shutdown margin assessment test, the wrong rod was accidentally pulled. This rod was assumed to be the maximum worth rod.

  RESULTS:

  - No criticality was calculated
  - However, for three regions, Keff's of about 0.99 were calculated (at 155 EFPD)
  - The number of regions that cause this problem decrease with core burnup

  ACTION:

  - Enhance core safety by providing total assurance that there will be no power to these rod drives, thus making the accident incredible

ACCIDENT ANALYSES

- Earthquakes
- Reactivity accidents
  - Excessive removal of poison
  - Loss of fission product poison
  - Core rearrangement
  - Introduction of steam to core
  - Sudden decrease reactor temperature
  - Rod withdrawal accidents
- Column deflection/misalignment
- Misplaced fuel element
- Coolant channel blockage
- Incidents involving electrical system
- Loss of cooling
- Leaks inside primary coolant system
  - Steam generators
  - Moisture ingress
- Fuel storage accidents

OVERALL SAR CONCLUSIONS

- No changes to the existing or interim Technical Specifications are required to proceed with defueling
- There are no unreviewed safety or environmental issues
- A Technical Specification change may be required toward the end of defueling concerning startup channel count rate and verification of Keff of less than 0.95 with all rods withdrawn

CONSERVATISMS

- An uncertainty error of 0.01 Delta K was assumed in a well understood core

- Remain shutdown by at least 0.01 Delta K even if a rod withdrawal accident is experienced with highest worth rod

  - Except for Region 33 at 200 EFPD

  - For all cases after 300 EFPD

- As core burnup continues the SAR calculations of shutdown margin at 155 EFPD become even more conservative

- SAR calculations assume Region 1 Rod is withdrawn

COMPARISON

REFUELING	DEFUELING
PRELIMINARY SEQUENCE IS SELECTED (BOTH INNER AND OUTER REGIONS)	SAME (OUTER)
ADEQUACY OF SHUTDOWN MARGIN IS CONFIRMED BY CODE	SAME
SHUTDOWN MARGIN VERIFICATION RODS ARE SELECTED (0.01ΔK + NEW FUEL $\Delta\rho$ + TEMPERATURE DEFECT, LCO 3.1.6)	SAME (0.01ΔK + TEMPERATURE DEFECT)
SAR IS WRITTEN UNDER 50.59 AS BASIS FOR REFUELING	SAME
CHANGES TO SEQUENCE OR SHUTDOWN MARGIN VERIFICATION-RODS ARE RE-EVALUATED UNDER 50.59	SAME
NEW FUEL IS INSERTED (POSITIVE REACTIVITY)	BORONATED DEFUELING ELEMENTS (NEGATIVE REACTIVITY)

DEFUELING SEQUENCE CONTROL

- PSC has demonstrated that 50.59 adequately controls the sequence and changes to the sequence

- Interim Technical Specification LCO 3.1.4 already provides the appropriate shutdown margin requirements

- Interim Technical Specification Surveillance SR 4.1.6 already requires an analysis and verification test of the shutdown margin

- The requirement to have no more than two control rods simultaneously removed (Interim Specification LCO 3.1.6) is understood by PSC to apply to regions with fuel

DESIGN FEATURES

- Defueling elements are not explicitly discussed in Section 6, however, use of boronated elements is discussed

- Purpose of Section is to describe "Design Characteristics of Special Importance to Each of the Physical Barriers and to the Maintenance of Safety Margins Which Have Not Been Covered in Any Other Specifications"

- The 'safety margin' associated with defueling elements is the 0.01 Delta K shutdown margin, which is already addressed in current Technical Specifications

FUEL HANDLING/FUEL STORAGE TECHNICAL SPECIFICATIONS

- Existing Technical Specifications address fuel handling and fuel storage requirements

- A comparison of existing and TSUP Specifications does not identify any significant differences

CONCLUSIONS

- WE HAVE NO UNREVIEWED SAFETY ISSUES. FORT ST. VRAIN CAN BE SAFELY DEFUELED USING EXISTING TECHNICAL SPECIFICATIONS. THEREFORE, PSC HAS THE AUTHORITY UNDER 10 CFR 50.59 TO PROCEED WITH DEFUELING FORT ST. VRAIN

- PSC WILL SUBMIT THE FORT ST. VRAIN DEFUELING SAR FOR NRC INFORMATION

- NRC DEFUELING APPROVALS ARE, IN THE OPINION OF PSC, NOT REQUIRED.

PROPOSED ACTIONS

- SUBMIT FINAL DEFUELING SAR FOR NRC INFORMATION
- SET UP A JOINT WORKING MEETING TO RESOLVE REMAINING TECHNICAL ISSUES