ATTACHMENT 2

TO P-93046

PROPOSED CHANGES

2.0 DEFINITIONS

The defined terms in this section appear in capitalized type and are applicable throughout these Technical Specifications.

2.1 ACTIONS

ACTIONS shall be that part of a specification which prescribes Required Actions under designated conditions, which shall be completed within specified Completion Times.

2.2 ACTIVATED GRAPHITE BLOCKS

ACTIVATED GRAPHITE BLOCKS shall include all activated graphite components that were inside the PCRV when there was irradiated fuel in the core. Defueling elements are not considered ACTIVATED GRAPHITE BLOCKS.

2.3 BASES

The BASES shall summarize the reasons for the Limiting Conditions, Applicabilities, ACTIONS, and Surveillance Requirements. In accordance with 10 CFR 50.36, the BASES are not considered part of the Decommissioning Technical Specifications.

2.4 CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and with the required accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel, considering system design, including the sensors and alarm, interlock and/or trip functions, and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

2.5 CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during its 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.

3.1 BASES (Continued)

Reactor building confinement integrity is taken credit for in the Heavy Load Drop and the Loss of AC Power accident analyses, as described in Section 3.4 of the Decommissioning Plan. (Reference 1)

LC

The LC establishes the minimum conditions required to ensure that Reactor Building confinement integrity is maintained during applicable accident scenarios (i.e., Heavy Load Drop and/or Loss of AC Power). The LC requirements are consistent with the accident analysis assumptions, and the criteria used during plant operation. It should be noted that the Reactor Building overpressure protection system louvers may be open provided there are no activities in progress involving the physical handling of any ACTIVATED GRAPHITE BLOCKS. For example, the louvers may be open while ACTIVATED GRAPHITE BLOCKS are being dried or are in temporary storage within the Reactor Building, as long as they are not being moved, cut, or otherwise physically handled.

APPLICABILITY

The Reactor Building confinement integrity applicability is based on complying with the off-site dose requirements established in the 10 CFR 100 guidelines and the EPA Protective Action Guidelines in the event of a Heavy Load Drop accident and/or Loss of AC Power. However, the Reactor Building overpressure protection system louvers may be open provided there are no activities in progress involving the physical handling of any ACTIVATED GRAPHITE BLOCKS.

Consistent with the Accident Analyses, ACTIVATED GRAPHITE BLOCKS include all graphite components inside the PCRV except defueling elements. The defueling elements are not activated. In the event of a load drop accident involving defueling elements, the resultant doses are low enough that confinement integrity or ventilation are not required.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1	Verify Reactor Building pressure is subatmospheric	12 hours
SR 3.2.2	Verify pressure drop across each HEPA filter is less than 6 inches of water, with a flow rate of at least 17,100 cfm	Weekly
SR 3.2.3	Verify HEPA filter bank satisfies in-place penetration and bypass leakage test acceptance criteria of less than 0.05 percent, using test procedure guidance in Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Rev. 2, March 1978, with a flow rate of at least 17,100 cfm	18 months, after structural maintenance on the HEPA filter housing, or after each complete or partial replacement of a HEPA filter bank

3.2 BASES (Continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1

Verification that Reactor Building pressure is subatmospheric ensures that the confinement integrity is intact. The 12 hour surveillance frequency is more frequent than the operating technical specification requirements.

SR 3.2.2

A pressure drop across the HEPA filter of less than 6 inches of water gauge at 90% of the filter design flow rate will indicate that the filters are not clogged by excessive amounts of foreign matter.

SR 3.2.3

Bypass leakage and penetration for High Efficiency Particulate Air (HEPA) filters are determined by dioctyl phthalate (DOP) testing. The filter penetration and bypass acceptance limits in the surveillances are applicable based on a HEPA filter efficiency of 99%, as assumed in the decommissioning accident analysis. The surveillance frequencies specified establish system performance capabilities.

Verification of the HEPA filter functions ensures system performance capabilities. The surveillance frequency is the same as the operating technical specifications.

REFERENCES

1. FSV Decommissioning Plan

2. Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, EPA-520/1-75-001-A, January 1990, U.S. Environmental Protection Agency

ATTACHMENT 3

TO P-93046

NO SIGNIFICANT HAZARDS
CONSIDERATION EVALUATION

DECOMMISSIONING OF THE FORT ST. VRAIN
NUCLEAR GENERATING STATION

10 CFR 50.92 EVALUATION

INTRODUCTION

The Decommissioning Order approving the Decommissioning Plan (DP) and authorizing the decommissioning of the Fort St. Vrain Nuclear Generating Station (FSV) was issued by the Nuclear Regulatory Commission (NRC) on November 23, 1992. As identified in the Decommissioning Order, decommissioning of the FSV facility is authorized in accordance with the DP (Reference 1) subject to several conditions. Condition (c) states that "If the licensee desires (1) a change in the TS or (2) to make a change in the facility or procedures described in the Decommissioning Plan or to conduct tests or experiments not described in the Decommissioning Plan, which involve an unreviewed safety question or a change in the TS, it shall submit an application for amendment of its license pursuant to 10 CFR 50.90 or request approval of a revision to the Decommissioning Plan."

Pursuant to 10 CFR 50.92, each application for amendment to a license must be reviewed to determine if the proposed change involves a significant hazards consideration. The Commission has provided standards for determining whether a significant hazards consideration exists [10 CFR 50.92(c)]. A proposed amendment to a license for a facility involves no significant hazards consideration if the change to 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.

This proposed amendment addresses the following subjects:

1. Revising Decommissioning Technical Specifications (DTS) SR 3.2.3 High Efficiency Particulate Air (HEPA) filter bank leakage test acceptance criteria of "less than 1 percent" to "less than 0.05 percent." Additionally, DTS SR 3.2.3 BASES will be revised to have a stated HEPA filtration efficiency of 99 percent in lieu of 95 percent.

2. Expanding the definition of ACTIVATED GRAPHITE BLOCKS to include all activated PCRV graphite components that were inside the PCRV when there was irradiated fuel in the core. Defueling elements are not considered Activated Graphite Blocks.

The proposed amendment as defined above has been reviewed and deemed not to involve a significant hazards consideration based on the following evaluation.

BACKGROUND

Section 3.4 of the NRC issued Safety Evaluation Report for FSV decommissioning states, "The results of the accident scenarios postulated for FSV decommissioning indicate radiation exposures to the general public are very low. The resulting analysis show that the radiological consequences at the EPZ [Emergency Planning Zone] are within the 10 CFR Part 100 guidelines and are only a small fraction of the EPA [Environmental Protection Agency] Protective Action Guidelines (EPA-520/1-75-001-A) and would therefore require no offsite response to the accident."

Section 3.4.5 of the DP postulated the dropping of the largest (2030 lbs.) large side reflector block. It was originally planned to section these reflector blocks into smaller pieces for packaging in LSA shipping containers. The analysis conservatively assumed that a shipping container containing the largest intact reflector block falls approximately 100 feet to the truck bay. It was conservatively assumed that one percent of the activity of the largest reflector block is dispersed from the drop. The dust is postulated to remain airborne and is assumed to escape the immediate area through the Reactor Building ventilation exhaust. Credit was taken for 95 percent removal efficiency for particulates afforded by the Reactor Building ventilation system. The radiological consequences to the general public from a postulated drop of an activated reflector block are well below the lowest level EPA Protective Action Guideline of one rem whole body dose and five rem to any specific organ. Therefore, no offsite emergency response actions are required.

Engineering assessments are being conducted to determine the most efficient method for packaging and disposal of the graphite components. The preliminary results of the evaluation indicate that occupational radiation exposures can be minimized by packaging more than one reflector block in burial liners and thus minimizing handling activities required by workers. However, to support alternate packaging schemes, the heavy load drop accident analysis must take credit for 99 percent HEPA filter efficiency to ensure offsite dose consequences remain a small fraction of the EPA Protective Action Guidelines. The specific methods used for packaging graphite components will be evaluated to confirm that they do not involve an unreviewed safety question as defined in 10 CFR 50.59. Furthermore, the evaluation will be performed to demonstrate that the radiological consequences are bounded by the doses of 121 millirem to the whole body and 215 millirem to the lung predicted for the worst case decommissioning accident of a postulated fire as presented in Section 3.4.6 of the DP.

The two changes to the DTS proposed in this submittal will allow various multiple block graphite packaging alternatives, and will allow for the possibility that block activity levels could be higher than assumed in the Activation Analysis, as follows:

1. Revising SR 3.2.3 to impose a more restrictive HEPA filter leak test criteria will allow the heavy load drop analysis to take credit for a higher (99% vs 95%) HEPA filter removal efficiency.

2. Revising the definition of Activated Graphite Blocks to include all remaining activated in-core graphite components effectively expands the applicability of LC 3.1 and LC 3.2, Reactor Building Confinement Integrity and Ventilation Exhaust System, respectively, to include core support blocks and core support posts. This allows the heavy load drop accident analyses for core support blocks and core support posts to take credit for Reactor Building confinement and for a 99% HEPA filter removal efficiency.

EVALUATIONS

I. HEPA Leakage Test Acceptance Criteria and Efficiency

Revision of the Reactor Building Ventilation System HEPA filter leak test acceptance criteria is consistent with:

- The HEPA filters' specified capability
- Regulatory Guide 1.52 (Reference 2)
- The Westinghouse Standard Technical Specifications (both NUREG-0452 and NUREG-1431)
- Decommissioning accident analysis assumptions used in NUREG/CR-0672, "Technology, Safety, and Costs of Decommissioning a Reference Boiling Water Reactor Power Station"

PSC originally chose to take credit for a HEPA filter efficiency of 95 percent in the decommissioning accident analysis, as reflected in the Basis for DTS LC 3.2. Use of a 95 percent HEPA filter efficiency in the decommissioning accident analysis was consistent with the assumptions in the original FSV Final Safety Analysis Report (FSAR) and a one percent HEPA filter leak test acceptance criteria in the DTS was based on the guidance provided in NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors" (STS). The Westinghouse STS stated that a one percent HEPA filter leak test criterion is applicable when a 95 percent HEPA filter efficiency is assumed in the NRC staff's safety evaluation, and a 0.05 percent leak test criterion is applicable when a 99 percent HEPA filter efficiency is assumed. The NRC's 1972 safety evaluation was consistent with Section 14.12.3 of the FSAR, and assumed a 95 percent efficiency.

As stated in Section 2.2.3.13 of the DP, the particulate filters have a specified and tested capability for removal of 99.9 percent of particulates that are 0.3 microns and greater in size and the HEPA filters conform to the applicable requirements of the following documents:

1. Military Specification MIL-F-51068 (D)
2. Underwriters Laboratories Standard UL-568-1977
3. Regulatory Guide 1.140
4. Regulatory Guide 1.52, Revision 2
5. ANSI Standard N509-1976

Each of the HEPA filter elements in each filter unit, manufactured from fiberglass conforming to Military Specification MIL-F-51079, is designed for a flow rate of 1200 CFM with a maximum initial (clean) pressure drop of 1.0 inch water gage. The design specifications require that the maximum penetration is 0.03 percent when tested with thermally generated monodispersed dioctylphthalate (DOP) smoke having a light scattering geometric mean droplet diameter of 0.3 microns. To ensure conformance to the design specifications, the HEPA filters are flow, leak and penetration tested by both the manufacturer and in-place by PSC using the test procedure guidance provided in Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, as specified in DTS SR 3.2.3. Although, DTS 3.2.3 specifies an in-place penetration and bypass leakage test acceptance criteria of less than 1 percent, recent in-place testing demonstrates that all HEPA banks meet the in-place leak test criteria of a penetration of less than 0.05 percent for HEPA filters.

Regulatory Guide 1.52 provides guidelines for the design, testing, and maintenance of air filtration and adsorption units of atmosphere cleanup systems designed to mitigate the consequences of postulated accidents. Paragraph C.5.c of Regulatory Guide 1.52 recommends an in-place leak test to confirm a penetration of less than 0.05 percent for HEPA filters. The Regulatory Guide permits the use of a 99 percent removal efficiency for particulates in accident dose evaluations provided that testing satisfies the in-place leak test criterion provided in the Regulatory Guide.

PSC is proposing to change the DTS SR 3.2.3 leakage test acceptance criteria of "less than 1 percent" to "less than 0.05 percent," consistent with the recommendations of Position C.5.c of Regulatory Guide 1.52. Additionally, DTS SR 3.2.3 BASES will be revised to reflect a HEPA filter efficiency of 99 percent in lieu of 95 percent.

In addition, revising the leakage test acceptance criteria in DTS SR 3.2.3 to "less than 0.05 percent" is consistent with the recommendation of NUREG-1431 (Revision 0), "Standard Technical Specifications, Westinghouse Plants," September 1992.

NRC guidance for developing the accident analysis sections of Decommissioning Plans is provided in the Draft Regulatory Guide entitled "Standard Format and Content For Decommissioning Plans For Nuclear Reactors", dated September 1989, Task DG-1005. Section 3 of this document states "References 2, 3, 4, and 5 contain information that may be helpful in

analyzing accidents during decommissioning." These references are NUREGs that provide detailed information on the technology, safety and costs of decommissioning a reference PWR, BWR, multiple reactor stations, and nuclear research and test reactors prepared by Pacific Northwest Laboratory. The accident analysis section of these references for a PWR, BWR and research reactor assumes a HEPA filter efficiency of 99.95 percent. For instance, Reference 3 of DG-1005 states the following in regards to accident analysis (Section N):

> To ensure proper air flow for the decommissioning workers, and to protect against uncontrolled atmospheric releases of radioactivity, HEPA filters are assumed to be installed in the Reactor Building, Turbine Generator Building, and Radwaste and Control Building exhaust ventilation systems. These filters are tested in-place on a regular basis. The measured particle collection efficiency of these HEPA filters is 99.95%. Airborne releases of radioactivity are assumed to take place at ground level for atmospheric dispersion calculations, and are assumed to pass through a single HEPA filter system with an assumed transmission factor of 5×10^{-4}.
>
> [Reference 3: H.D. Oak et al., "Technology, Safety, and Costs of Decommissioning a Reference Boiling Water Reactor Power Station" (prepared for the Nuclear Regulatory Commission by Pacific Northwest Laboratory), NUREG/CR-0672, June 1980.]

Adoption of a 99 percent filter efficiency would be a factor of 20 times more conservative than that assumed in the accident analyses for the reference reactors.

II. Definition of ACTIVATED GRAPHITE BLOCKS

The intent of this definition is related to DTS LCs 3.1 and 3.2, which specify the requirements needed to ensure Reactor Building confinement integrity, subatmospheric conditions, and operability of the Reactor Building ventilation exhaust fans and filters. Heavy load drops of graphite components are not anticipated during decommissioning. However, should such load drops occur, DTS 3.1 and 3.2 provide assurance that activity releases will be filtered, as necessary, such that dose consequences do not exceed a small fraction of EPA Protective Action Guidelines.

By expanding the definition of ACTIVATED GRAPHITE BLOCKS, the applicabilities of LCs 3.1 and 3.2 are implicitly being expanded to include all graphite components except defueling elements. To accommodate various possible graphite component packaging schemes and to allow for potentially higher activity levels in the core support blocks and core support posts than were predicted by the Activation Analysis, PSC considers that LCs 3.1 and 3.2 should apply to activities involving handling of the core support blocks and core support posts. Therefore, extending the applicability of the DTS for Reactor Building confinement integrity and ventilation exhaust fan and filter operability ensures that the requirements are applied during all activities involving activated graphite components. This ensures that potential off-site consequences from

accident scenarios postulated for FSV decommissioning will be very low. In all cases, the radiological consequences at the EPZ will be within the 10 CFR Part 100 guidelines and will be a small fraction of the EPA Protective Action Guidelines.

CONCLUSIONS

Based on the information presented above, the following conclusions can be reached with respect to 10 CFR 50.92.

1. Revising DTS SR 3.2.3 leakage test acceptance criteria to "less than 0.05 percent," and DTS SR 3.2.3 BASES to have a stated HEPA filter efficiency of 99 percent does not increase the probability or consequences of an accident previously evaluated in the DP. Revising the leakage test acceptance criteria, specified in DTS SR 3.2.3, to "less than 0.05 percent," will make the DTS consistent with the recommendations of Position C.5.c of Regulatory Guide 1.52 and consistent with the recommendation of NUREG-1431 (Revision 0), "Standard Technical Specifications, Westinghouse Plants," September 1992.

   The integrity of the Reactor Building, in conjunction with operation of the Reactor Building ventilation system, limits the off-site doses under normal and abnormal conditions during decommissioning activities. Changing the leakage test acceptance criteria does not create any new failure modes for the ventilation system or the integrity of the Reactor Building confinement. No new limiting single failure has been identified for the HEPAs. The ability of the Reactor Building ventilation exhaust system to perform its filtering function and the integrity of the Reactor Building confinement are not adversely affected by the proposed changes. Furthermore, the HEPAs and leakage testing are not initiators for any of the postulated DP accidents analyzed. Therefore, it can be concluded that revising DTS SR 3.2.3 leakage test acceptance criteria to "less than 0.05 percent," and DTS SR 3.2.3 BASES to have a stated HEPA filter efficiency of 99 percent has no effect on the probability of occurrence of any accident evaluated in the DP.

   Expanding the DTS 2.2 definition of ACTIVATED GRAPHITE BLOCKS to include all activated graphite components in the PCRV implicitly extends the applicability of the DTS for Reactor Building confinement integrity and ventilation exhaust fan and filter operability to activities involving essentially all graphite components removed from the PCRV and remaining inside the Reactor Building. This proposed change is not expected to create any new limiting single failure modes for the ventilation system or the Reactor Building confinement.

   With respect to the consequences of accident analyses, analyses of postulated decommissioning accidents are provided in Section 3.4 of the DP. The Heavy Load Drop accident, in Section 3.4.5 of the DP, is the only accident analysis that takes credit

for filtration of air released from the Reactor Building. Changing the DTS SR 3.2.3 leakage test acceptance criteria to "less than 0.05 percent," and subsequently changing DTS SR 3.2.3 BASES to have a stated HEPA filter efficiency of 99 percent will decrease the offsite radiological consequences of the heavy load drop accident involving a single large side reflector block as reported in the DP. Furthermore, the proposed change will make the test acceptance criteria more stringent and will therefore provide greater assurance that the radiological consequences from the postulated decommissioning accident scenarios remain well within the 10 CFR Part 100 guidelines and are only a small fraction of the EPA Protective Action Guidelines.

In all cases, including the postulated dropping of multiple large side reflector blocks, the radiological consequences from the postulated decommissioning accidents will be bounded by the doses of 121 millirem to the whole body and 215 millirem to the lung predicted for the worst case decommissioning accident of a postulated fire, as presented in Section 3.4.6 of the DP. The integrity of the Reactor Building, in conjunction with operation of the ventilation exhaust system, will continue to limit the off-site doses under normal and abnormal conditions during decommissioning activities.

Expanding the DTS 2.2 definition of ACTIVATED GRAPHITE BLOCKS to include all activated graphite components in the PCRV extends the applicability of the DTS for Reactor Building confinement integrity and ventilation exhaust fan and filter operability to activities involving core support blocks and core support posts. These graphite components have lower activity concentrations than the large side reflectors used in the heavy load drop accident analysis, and their packages will be evaluated to ensure that the consequences of a postulated drop of core support blocks and posts would be bounded by the radiological consequences predicted for the dropping of large side reflectors. In all cases, the radiological consequences at the EPZ will be a small fraction of the EPA Protective Action Guidelines.

2. Revising DTS SR 3.2.3 leakage test acceptance criteria to "less than 0.05 percent," and DTS SR 3.2.3 BASES to have a stated HEPA filter efficiency of 99 percent does not create the possibility of different types of accidents or malfunctions other than those evaluated previously in the DP. Revising the leakage test acceptance criteria, specified in DTS SR 3.2.3, to make the Decommissioning Technical Specifications consistent with the recommendations of Position C.5.c of Regulatory Guide 1.52 and consistent with Technical Specifications of light water cooled commercial nuclear power plants does not place the ventilation system in configurations conducive to the occurrence of accidents or malfunctions not previously evaluated.

As previously stated, expanding the DTS 2.2 definition of ACTIVATED GRAPHITE BLOCKS to include all graphite components in the PCRV, except for defueling elements, extends the applicability of the DTS for Reactor Building confinement integrity and ventilation exhaust fan and filter operability to activities involving essentially all remaining in-core graphite components. However, no new performance requirements are

[illegible]

being imposed on the ventilation system or its components such that any design criteria is expected to be exceeded. Therefore, the original design intent and performance criteria of the ventilation system continue to be met.

3. Revising the leakage test acceptance criteria to "less than 0.05 percent," will allow use of 99 percent filter efficiency in the heavy load drop accident, consistent with the recommendations of Position C.5.c of Regulatory Guide 1.52 and with the recommendation of NUREG-1431 (Revision 0), "Standard Technical Specifications, Westinghouse Plants," September 1992.

   Although this represents a change to an assumption used in the Heavy Load Drop accident, it does not involve a significant reduction in a margin of safety. In all cases, the radiological consequences from the postulated decommissioning accident scenarios will remain a small fraction of the one rem whole body dose and five rem to any specific organ guidelines cited in the EPA Protective Action Guidelines.

   Expanding the DTS 2.2 definition of ACTIVATED GRAPHITE BLOCKS to include all graphite components in the PCRV except defueling elements, ensures that potential off-site consequences from accident scenarios postulated for handling of multiple core support blocks and/or core support posts will be very low. In all cases, the radiological consequences at the EPZ from postulated decommissioning accidents will be a small fraction of the EPA Protective Action Guidelines. The potential offsite radiological consequences from any accident involving graphite components will remain within the bounds of safe, analyzed conditions as defined in the DP. As such, the margin of safety, as defined in the Bases to the Decommissioning Technical Specifications will not be reduced.

Based upon the preceding evaluation, it has been determined that the proposed changes revising DTS SR 3.2.3 leakage test acceptance criteria to "less than 0.05 percent," revising DTS SR 3.2.3 BASES to have a stated HEPA filter efficiency of 99 percent, and extending the definition of ACTIVATED GRAPHITE BLOCKS to include all graphite components in the PCRV, except defueling elements, do not involve a significant increase in the probability or consequences of an accident or malfunction previously evaluated, create the possibility of a new or different kind of accident or malfunction from any previously evaluated or involve a significant reduction in a margin of safety in the Basis of any Technical Specification. Therefore, it is concluded that the licensing amendment does not involve a Significant Hazards Consideration as defined in 10 CFR 50.92 (c).

REFERENCES

1. Fort St. Vrain [Proposed] Decommissioning Plan, dated April 17, 1992.

2. Regulatory Guide 1.52, Revision 2, dated March 1978, "Design Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."