DECOMMISSIONING

TECHNICAL SPECIFICATIONS

For

FORT ST. VRAIN

Unit No. 1

Docket No. 50-267

Appendix A
to
Facility License No. DPR-34

FORT ST. VRAIN

DECOMMISSIONING TECHNICAL SPECIFICATIONS

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## 1.0 INTRODUCTION

These Decommissioning Technical Specifications are applicable during the decommissioning of the Fort St. Vrain (FSV) reactor. Decommissioning is considered to begin after all of the nuclear fuel has been removed from the FSV Reactor Building and after the NRC has approved the Decommissioning Plan.

The Fort St. Vrain Nuclear Generating Station originally operated as a High Temperature Gas-Cooled Reactor, which supplied steam to a turbine generator. The facility may be converted to utilize a gas-fired boiler. Although some of the balance of plant systems will be retained for use after the conversion, many plant systems have been taken out of service and are not described in these Decommissioning Technical Specifications.

Activities that will be undertaken in accordance with these Decommissioning Technical Specifications include the dismantlement and decommissioning (DECON) of the radiologically activated and contaminated portions of the facility to release all site areas for unrestricted use.

There are two categories of FSV Technical Specifications:

- "Decommissioning Technical Specifications (DTS)" include Amendment __ and all subsequent amendments.

- "Operating Technical Specifications" refers to the historical Technical Specifications included in all previous amendments.

# 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 the reflector blocks and spacer blocks. Other graphite items, such as defueling elements, core support blocks, and core support posts, 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.

DEFINITIONS (Continued)

2.6 CHANNEL FUNCTIONAL TEST

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

2.7 EXCLUSION AREA BOUNDARY

The EXCLUSION AREA BOUNDARY shall enclose the decommissioning Emergency Planning Zone (EPZ), as shown on Figure 4.1. The EXCLUSION AREA BOUNDARY is a minimum of 100 meters from the Reactor Building, Fuel Storage Building, and Radioactive Waste Compactor Building.

2.8 MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with decommissioning the plant. Individuals who are occupationally associated with the conversion of the plant, and persons who enter the site to service equipment or make deliveries, are included in this category. MEMBER(S) OF THE PUBLIC also includes persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

2.9 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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 Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 5.4.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 5.5.1 and 5.5.2.

DEFINITIONS (Continued)

2.10 OPERABLE - OPERABILITY

A component or system shall be OPERABLE or have OPERABILITY when it is capable of performing its intended safety function within the required range. The component or system shall be considered OPERABLE when: (1) it satisfies the Limiting Conditions defined in these Decommissioning Technical Specifications, and (2) it has been satisfactorily tested periodically in accordance with the Surveillance Requirements defined in these Decommissioning Technical Specifications.

2.11 PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the procedure, current formulas, sampling, analyses, tests, 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.

2.12 RADIATION SAFETY

RADIATION SAFETY shall refer to activities involving the final release of previously contaminated or activated site structures, systems, components, or materials for unrestricted use; to activities that could result in exposures to project personnel or the public in excess of 10 CFR 20 limits; and to activities involving packaging and transportation of radioactive material.

2.13 UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area inside or outside the EXCLUSION AREA BOUNDARY (or Emergency Planning Zone) to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

## 3.0 GENERAL REQUIREMENTS

3.0.1 Compliance with the Limiting Conditions (LC) contained in Section 3 of the Specifications is required during Fort St. Vrain Decommissioning; except that upon discovery of a failure to meet the LC, the associated Required Actions shall be met within the specified Completion Times.

3.0.2 Noncompliance with a specification shall exist when the requirements of the LC and associated Required Actions are not met within the specified Completion Times. If the LC is restored prior to expiration of the specified Completion Time, the Required Actions need not be completed.

3.0.3 Surveillance Requirements shall be met as specified in the Applicability for individual LCs unless otherwise stated in an individual Surveillance Requirement. Failure to meet a Surveillance Requirement, except as provided in 3.0.4, shall constitute failure to meet the LC. Surveillance Requirements do not have to be performed on inoperable equipment.

3.0.4 Each Surveillance Requirement, any Required Actions which require the performance of a Surveillance Requirement, and any Required Action with a Completion Time requiring the periodic performance of an action on a "once per..." interval, shall be performed within the specified Frequency with a maximum allowable extension not to exceed 25% of the time interval.

3.0.5 For a Surveillance Requirement not performed within the Frequency defined by 3.0.4, the ACTIONS are applicable at the time it is identified that the Surveillance has not been performed. The Required Actions may be delayed for up to 24 hours to permit the completion of the Surveillance when the Completion Time of the Required Action is less than 24 hours. When a Surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the Completion Times of the ACTIONS are applicable at that time.

## 3.0 BASES

3.0.1 and 3.0.2 — 3.0.1 and 3.0.2 establish the general requirements applicable to LCs. These requirements are based on the requirements consistent with operating plants' Limiting Conditions for Operation per the Code of Federal Regulations, 10 CFR 50.36 (c)(2). 3.0.1 establishes the Applicability statement within individual specifications as the requirement for when conformance to the LC is required for safe decommissioning of the unit. The Required Actions establish those remedial measures that must be taken within specified Completion Times when requirements of a LC are not met.

3.0.2 establishes that noncompliance with a specification exists when the requirements of the LC are not met and the associated Required Actions have not been met within the specified Completion Times. The purpose of this general requirement is to clarify that: (1) completion of the Required Actions within the specified Completion Times constitutes compliance with a specification, and (2) completion of the remedial measures of the Required Actions is not required when compliance with an LC is restored within the Completion Time specified in the associated ACTIONS, unless otherwise specified.

3.0.3 - 3.0.5 — 3.0.3, 3.0.4, and 3.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the requirements consistent with operating plants' Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36 (c)(3).

3.0.3 establishes the requirements that Surveillance Requirements must be met during the conditions specified in the Applicability for which the requirements of the LC apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this general requirement is to ensure that Surveillances are performed to verify the status of systems and components and that parameters are within specified limits. Surveillance Requirements do not have to be performed when outside of the Applicability of the LC unless otherwise specified.

3.0 BASES (Continued)

3.0.4 establishes the conditions under which the specified Frequency for Surveillance Requirements, Required Actions which require the performance of a specific Surveillance Requirement, and any Required Action with a Completion Time requiring the periodic performance of an action on a "once per ..." interval may be extended. 3.0.4 permits an extension of the Frequency to facilitate Surveillance scheduling and consideration of decommissioning conditions that may not be suitable for conducting the Surveillance; e.g., maintenance activities.

The limit of 3.0.4 is based on engineering judgement and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured throughout Surveillance activities is not significantly degraded beyond that obtained from the specified Surveillance Frequency.

3.0.5 establishes that the failure to perform a Surveillance within the allowed Surveillance Frequency, defined by the provisions of 3.0.4, is a condition that constitutes a failure to meet the OPERABILITY requirements for an LC. Under the provisions of this general requirement, systems and components are assumed to be OPERABLE when the associated Surveillance Requirements have not been met. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirement frequency. This general requirement also clarifies that the ACTIONS are applicable when Surveillances have not been completed within the allowed Surveillance Frequency and that the Completion Times of the Required Actions apply from the point in time it is identified that a Surveillance has not been performed and not at the time that the allowed Surveillance Frequency was exceeded.

3.0 BASES (Continued)

If the Completion Times of the ACTIONS are less than 24 hours, a 24-hour allowance is provided to permit a delay in implementing the Required Actions. This provides adequate time to complete Surveillance Requirements that have not been performed. If a Surveillance is not completed within the 24-hour allowance, the Completion Times of the ACTIONS are applicable at that time.

For the purpose of making the transition from the operating Technical Specifications to the Decommissioning Technical Specifications, surveillances performed under the operating Technical Specifications may be utilized to satisfy the applicable surveillance requirements of the Decommissioning Technical Specifications.

## 3.1 REACTOR BUILDING CONFINEMENT INTEGRITY

LC 3.1 Reactor Building confinement integrity shall be maintained with:

a. The Reactor Building overpressure protection system louvers closed*, and

b. Either:

1. The outer truck bay closures closed, or

2. The inner truck bay closures closed.

APPLICABILITY: Whenever ACTIVATED GRAPHITE BLOCKS have been removed from the PCRV shielding water and remain inside the Reactor Building*

ACTION

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Do not have Reactor Building confinement integrity | A.1 Suspend activities involving physical handling of ACTIVATED GRAPHITE BLOCKS within the Reactor Building | 1 hour |

* 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.

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.1 | Verify all Reactor Building overpressure protection system louvers are in the closed position, except as permitted by LC 3.1. | Daily, when physical handling of ACTIVATED GRAPHITE BLOCKS is in progress |
| SR 3.1.2 | Verify inner truck bay closures are closed | Prior to opening outer truck bay closures |

## 3.1 BASES

BACKGROUND

The integrity of the Reactor Building, in conjunction with operation of the ventilation exhaust system, limits the off-site doses under normal and abnormal conditions during decommissioning activities. In the unlikely event of a major release of activity from the Prestressed Concrete Reactor Vessel (PCRV) dismantlement (i.e., Heavy Load Drop Accident), the combination of the Reactor Building integrity and ventilation exhaust system would act to keep off-site doses well below 10 CFR 100 guidelines and within a small fraction of EPA guidelines (Reference 2).

The integrity of the Reactor Building confinement is normally maintained with the exterior closures and the overpressure protection system louvers closed. The truck bay includes two redundant sets of closures. The outer closures have historically included a truck door and the personnel access door in the truck door. The inner closures have historically included the truck bay floor hatch, the truck bay overhead sliding hatch, and the internal personnel door. During decommissioning, there will continue to be two redundant closures which may include the addition of new outer truck doors, external to the original truck doors, in an airlock-type configuration.

The Reactor Building shall be maintained subatmospheric at all times including normal access (see LC 3.2). Subatmospheric conditions can be maintained with several louver banks open. The overpressure protection system louvers may be opened on a controlled basis for various reasons (e.g., to provide extra ventilation cooling during hot weather).

The inner closures of the truck bay are closed to ensure integrity of the Reactor Building confinement prior to the opening of the outer truck doors to the truck bay.

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 reflector blocks and spacer blocks as defined in Specification 2.2. The activation level of other graphite materials is significantly less than the reflector blocks and spacer blocks. In the event of a load drop accident involving other graphite materials, the resultant doses are low enough that confinement integrity or ventilation are not required.

3.1 BASES (Continued)

ACTIONS

A.1

When Reactor Building confinement integrity is breached, suspend activities involving physical handling of ACTIVATED GRAPHITE BLOCKS within the Reactor Building. The 1 hour completion time to suspend physical handling of the ACTIVATED GRAPHITE BLOCKS allows an orderly suspension of activities.

SURVEILLANCE REQUIREMENTS

SR 3.1.1

The Reactor Building overpressure protection system louvers are verified in their closed position daily during activities when they are required to be closed, that is, during, although not necessarily contemporaneously with, physical handling of ACTIVATED GRAPHITE BLOCKS.

SR 3.1.2

Prior to opening the outer truck bay closures, the inner truck bay closures are verified closed. While the outer truck bay closures are open, locks or signs are posted on the inner truck bay closures to prevent them from being opened. This ensures Reactor Building confinement integrity.

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

## 3.2 REACTOR BUILDING VENTILATION EXHAUST SYSTEM

LC 3.2 The Reactor Building ventilation exhaust system shall be OPERABLE with:

a. Reactor Building internal pressure subatmospheric, and

b. At least one of the three ventilation exhaust trains OPERABLE, with each train consisting of one exhaust fan (C-7301, C-7302, or C-7302S) and the HEPA filter section of the associated filter assembly (F-7301, F-7302, or F-7302S).

APPLICABILITY: Whenever ACTIVATED GRAPHITE BLOCKS have been removed from the PCRV shielding water and remain inside the Reactor Building

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Reactor Building pressure is atmospheric or greater | A.1 Suspend activities involving physical handling of ACTIVATED GRAPHITE BLOCKS within the Reactor Building | 1 hour |
| B. All exhaust trains inoperable | B.1 Restore at least one ventilation exhaust train to OPERABLE status | 12 hours |
| C. Required Action B.1 not met within Completion Time | C.1 Suspend activities involving physical handling of ACTIVATED GRAPHITE BLOCKS within the Reactor Building | 12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.1 | Verify Reactor Building pressure is subatmosphe. c | 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 1 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

BACKGROUND

The Reactor Building ventilation exhaust filter system is designed to filter the Reactor Building atmosphere prior to release to the vent stack during both normal and most accident conditions during decommissioning.

The system consists of three trains, one of which is normally in continuous operation. The design flow rate for each train is 19,000 cfm. Allowing 10% for degradation, the minimum flow rate is 17,100 cfm. One train is sufficient to maintain the Reactor Building subatmospheric and thereby minimize unfiltered fission product release from the building. With only one exhaust fan operating, the ventilation system controls will throttle fresh air supply to the air handler in order to reduce the pressure.

The Reactor Building is maintained in a subatmospheric condition to ensure that all air leakage will be inward and to minimize unfiltered fission product release from the building. The ventilation system was designed to maintain a subatmospheric condition approximately 1/4 inch water gauge negative. In actual practice, the Reactor Building pressure is normally 0.15 to 0.20 inches water gauge negative, depending on building activities and ventilation system configuration. There is an alarm at approximately 0.08 inches water gauge negative, and the outside air supply will fully close if the building pressure increases to atmospheric.

The Reactor Building ventilation exhaust system is taken credit for in the Heavy Load Drop accident analysis, as described in Section 3.4 of the Decommissioning Plan (Reference 1).

LCs

The LC establishes the minimum conditions required to ensure the Reactor Building ventilation exhaust system is maintained while the potential exists for a drop of an ACTIVATED GRAPHITE BLOCK. One train is sufficient to maintain the Reactor Building subatmospheric and thereby minimize unfiltered fission product release from the building.

HEPA filters provide the required particulate filtration.

3.2 BASES (Continued)

APPLICABILITY

The Reactor Building ventilation exhaust system will remain OPERABLE, providing filtration of effluents to the environment, while the potential exists for dropping an ACTIVATED GRAPHITE BLOCK.

ACTIONS

A.1

When the Reactor Building pressure is atmospheric or greater, suspend activities involving physical handling of ACTIVATED GRAPHITE BLOCKS within the Reactor Building. The one hour completion time to suspend activities involving physical handling of ACTIVATED GRAPHITE BLOCKS within the Reactor Building minimizes the time exposure of the Reactor Building to atmospheric or greater conditions and is a conservative time frame. The suspension of physical handling activities is acceptable because all analyzed accidents assume something active is happening - no passive postulated accidents will result in radiological conditions where the need for ventilation and confinement exists.

B.1

The ability of the Reactor Building ventilation exhaust system to perform its filtering function during a Heavy Load (ACTIVATED GRAPHITE BLOCK) Drop is dependent on at least one exhaust train being OPERABLE. With all the exhaust trains inoperable, restore at least one ventilation exhaust train to OPERABLE status. A ventilation train may be operating but not OPERABLE, e.g., in the event a required Surveillance is not completed on time. In this case, a 12 hour completion time is reasonable since the Reactor Building will still be maintained at subatmospheric conditions.

C.1

When Required Action B.1 cannot be completed within the required Completion Time, all activities involving physical handling of ACTIVATED GRAPHITE BLOCKS within the Reactor Building are suspended. Twelve hours is reasonable to suspend handling activities. The suspension of physical handling activities is acceptable because all analyzed accidents assume something active is happening - no passive postulated accidents will result in radiological conditions where the need for ventilation and confinement exists.

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 95%. 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

## 3.3 RADIATION MONITORING INSTRUMENTATION

LC 3.3 The area radiation monitoring instrumentation channels shown in Table 3.3-1 shall be OPERABLE with their alarm setpoints within the limits specified for the activities in progress, depending on whether Radiation Work Permit (RWP) controls are in effect.

APPLICABILITY: At all times, until all significantly contaminated or activated items that could exceed alarm setpoints have been removed from the Reactor Building.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more radiation monitor channel alarm setpoint exceeds value in Table 3.3-1 | A.1 Adjust alarm setpoint within limit<br>OR<br>A.2 Declare the channel inoperable | 4 hours |
| B. One or more radiation monitor channels inoperable | B.1 Place a portable monitor (with alarm) in the area | 6 hours |

## SURVEILLANCE REQUIREMENTS

SR 3.3.1 Perform the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION surveillances as shown in Table 3.3-2.

TABLE 3.3-1

RADIATION MONITORING INSTRUMENTATION

| INSTRUMENT | ALARM SETPOINTS | |
|---|---|---|
| | DURING ACTIVITIES NOT CONTROLLED BY AN RWP | DURING ACTIVITIES CONTROLLED BY AN RWP |
| a. Refueling Floor | < 15 mR/hr | < 100 mR/hr* |
| b. Truck Bay | < 15 mR/hr | < 100 mR/hr* |

* Monitors may be reset to alarm at a radiation level within a factor of 2 of the expected radiation level.

TABLE 3.3-2

SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL FUNCTIONAL TEST | CHANNEL CALIBRATION |
|---|---|---|---|
| a. Refueling Floor | Daily | Monthly | 6 months |
| b. Truck Bay | Daily | Monthly | 6 months |

## 3.3 BASES

BACKGROUND

The radiation monitoring instrumentation required by this specification at all times during decommissioning activities, until all significantly contaminated or activated items that could exceed alarm setpoints have been removed from the Reactor building, includes two area radiation monitors, one on the refueling floor and one in the truck bay of the Reactor Building. These monitors serve as accident monitors to detect unplanned radiation levels in the Reactor Building, that should be investigated and appropriately resolved.

Decommissioning of Fort St. Vrain involves the removal of activated and contaminated material which inherently will result in increased radiation levels in the Reactor Building. These increased radiation levels will normally be anticipated and planned for, with monitoring provided as required. Individual work activities will be performed under Radiation Work Permits (RWPs), which will include monitoring provisions. Also, gaseous effluent releases will be monitored and controlled by the Offsite Dose Calculation Manual (ODCM) program. Liquid releases will also be monitored and controlled, in accordance with the ODCM program.

The monitors required by this specification are not relied upon in any accident analysis, but they are provided to detect abnormal conditions that could indicate unplanned or accidental radiation levels.

LC

The LC establishes the minimum conditions required to ensure the radiation levels are measured in the area served by the individual channels and that an alarm is initiated when the radiation level setpoint is exceeded.

Different alarm setpoints are allowable for the radiation monitors, depending on the activities in progress. While Radiation Work Permit (RWP) controls are in effect, a 100 mR/hr setpoint will detect unplanned radiation levels. This alarm setpoint may be raised during activities that are expected to exceed this setpoint, but no greater than a factor of 2 of the expected radiation level. At all other times, an alarm setpoint of 15 mR/hr is specified. These alarm setpoints will avoid nuisance alarms while still providing for detection of unplanned radiation levels.

3.3 BASES (Continued)

APPLICABILITY

This LC is applicable at all times.

ACTIONS

A.1 or A.2

When one or more radiation monitor channel alarm/trip setpoint exceeds the values in Table 3.3-1 either adjust the alarm/trip setpoint within its limits or declare the channel inoperable. The Required Action and Completion Time of 4 hours is consistent and comparable with Standard Technical Specifications.

B.1

When one or more radiation monitor channels is inoperable, place a portable monitor with an alarm in the area. The OPERABILITY of the radiation monitoring channels ensures that the radiation levels are measured in the areas served by the individual channels and an alarm is initiated when the radiation level trip setpoint is exceeded. A 6 hour Completion Time is reasonable to complete the Required Action.

SURVEILLANCE REQUIREMENTS

SR 3.3.1

The surveillance requirements frequencies specified for CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION conform to industry practice and the surveillance frequencies given in Standard Technical Specifications and are adequate to ensure the proper operation of these detectors.

REFERENCES

1. FSV Decommissioning Plan
2. Offsite Dose Calculation Manual Program

## 3.4 PCRV SHIELDING WATER TRITIUM CONCENTRATION

LC 3.4 Tritium concentration in PCRV shielding water shall not exceed 62.4 µCi/cc.

APPLICABILITY: Whenever there is shielding water within the PCRV.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. PCRV shielding water tritium concentration > 62.4 µCi/cc | A.1 Reduce tritium concentration to ≤ 62.4 µCi/cc<br>OR<br>A.2 Perform engineering evaluation to verify total tritium content ≤ 1 E+5 Ci | 72 hours |
| B. Required Actions not met within required Completion Time | B.1 Prepare and submit to the NRC a Special Report describing the safety concerns and the plans for restoring tritium concentration to within the limit | The next 30 days |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.1 | Verify PCRV shielding water tritium concentration within limits | Daily, during initial filling of the PCRV with shielding water, until tritium concentration is less than 0.1 μCi/cc for three consecutive samples. |
| SR 3.4.2 | Verify PCRV shielding water tritium concentration within limits | Weekly, after tritium concentration is less than 0.1 μCi/cc, until tritium concentration is less than 0.01 μCi/cc for three consecutive samples. |
| SR 3.4.3 | Verify PCRV shielding water tritium concentration within limits | Monthly, after tritium concentration is less than 0.01 μCi/cc |

## 3.4 BASES

BACKGROUND

During Decommissioning of Fort St. Vrain, the Prestressed Concrete Reactor Vessel (PCRV) cavity will be flooded with water to facilitate the removal of the reactor core components. PCRV dismantlement activities will begin only after all spent fuel has been removed from the reactor building. The water will be circulated, and purified by the PCRV water circulation system to gradually decrease the radioactivity, except tritium, in the water. Thus, the flooding of the PCRV will provide shielding for the workers associated with PCRV dismantlement activities.

There are a number of systems associated with the flooding of the PCRV to control radioactive material. Their functions include filtration of the PCRV water inventory, partial demineralization for controlling dissolved solids, and "Feed and Bleed" for adding clean makeup water and for removing contaminated (primarily tritium) water. The initial fluctuating increase in the tritium concentration during the flooding of the PCRV will be controlled by the "Feed and Bleed" dilution process. In accordance with the ODCM, released tritiated water will normally be treated as normal liquid radwaste, diluted and released at a controlled rate.

A maximum PCRV shielding water tritium concentration is assumed in the Loss of PCRV Shielding Water accident analysis, as described in Section 3.4 of the Decommissioning Plan (Reference 1).

For this analysis, it is conservatively assumed that the theoretical maximum amount of tritium is transferred to the PCRV shielding water from the graphite blocks, which is approximately 1 E+5 Curies. The tritium concentration in the spilled water is calculated to be 62.4 μCi/cc.

3.4 BASES (Continued)

LC

The LC establishes the maximum concentration tolerable in the PCRV shielding water to ensure adequate protection to the MEMBERS OF THE PUBLIC. The LC requirements are consistent with the accident analysis assumptions. It should be noted that the accident analysis assumed 1 E+5 Curies released. The resulting tritium concentration of 62.4 µCi/cc was chosen as the LC requirement because it is easier to determine a tritium concentration for surveillance monitoring purposes.

APPLICABILITY

This LC is applicable whenever there is shielding water within the PCRV, until all ACTIVATED GRAPHITE BLOCKS have been removed from the PCRV, after which there is no credible source of additional tritium.

ACTIONS

A.1 or A.2

When the PCRV shielding water tritium concentration is greater than 62.4 µCi/cc it is prudent to either reduce the concentration to less than or equal to 62.4 µCi/cc or perform an engineering evaluation to verify that the total tritium content is less than or equal to 1 E+5 Curies. A completion time of 72 hours is a reasonable amount of time to change the concentration of large water volumes and to perform associated analyses.

B.1

When a Required Action cannot be completed within the required Completion Time, a Special Report must be prepared and submitted to the NRC describing the safety concerns and the plans for restoring tritium concentration to within its safety analysis limit. The preparation and submittal of a Special Report is an acceptable action because the 1 E+5 Curie analysis value results in doses far below the limits allowed by Reference 2. The Special Report will be prepared as described in Specification 5.5.4.

3.4 BASES (Continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1 and 3.4.2

Verification of PCRV shielding water tritium concentration limits ensures adequate protection to the MEMBERS OF THE PUBLIC. The daily surveillance frequency during the filling of the PCRV with the shielding water, until tritium concentration decreases below 0.1 µCi/cc, will detect any fluctuations in the tritium concentration during non-steady state conditions. Also, performing daily sampling until tritium concentration is less than 0.1 µCi/cc for three consecutive samples ensures that equilibrium conditions are achieved before the surveillance frequency is decreased.

The 7 day surveillance frequency will ensure that a fluctuation in the tritium concentration during subsequent material handling activities will be detected. After equilibrium tritium concentration has decreased below 0.01 µCi/cc, monthly sampling will be performed. This is conservative with respect to Regulatory Guide 8.32 sampling requirements.

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
3. Regulatory Guide 8.32, Criteria for Establishing a Tritium Bioassay Program.

# 4.0 DESIGN FEATURES

## 4.1 Site

The Fort St. Vrain Nuclear Generating Station is located approximately 35 miles north of Denver and 3.5 miles northwest of the town of Platteville, in Weld County, Colorado.

The site consists of 2798 acres. The EXCLUSION AREA BOUNDARY encloses the decommissioning Emergency Planning Zone, as shown on Figure 4-1.

Points where radioactive gaseous and liquid effluents are released are shown on Figure 4-1.

## 4.2 Reactor Building

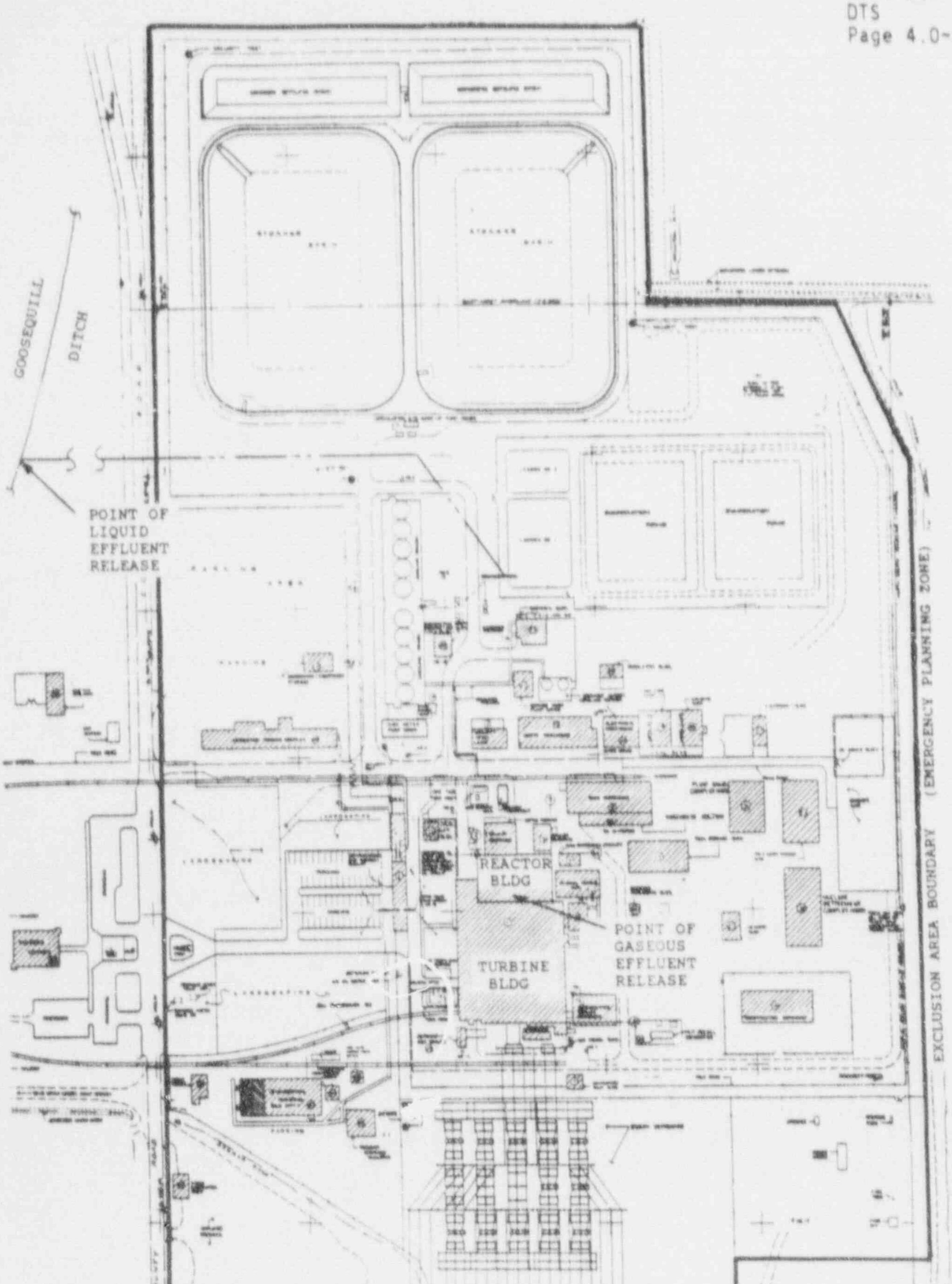
The Reactor Building houses the prestressed concrete reactor vessel (PCRV), fuel handling area, fuel storage wells, fuel shipment preparation facilities, decontamination and radioactive liquid and gas waste processing equipment, and most reactor plant process and service systems.

Decommissioning will not involve any major modifications to the Reactor Building structural steel without verification of the seismic qualification, as described in Section 2.2.1 of the Decommissioning Plan.

## 4.3 PCRV Water Leakage Prevention

The PCRV will be filled with water to provide shielding for workers during initial PCRV internal dismantlement activities. To prevent leakage from the PCRV, all penetrations which are below the PCRV water line and have had their instrumentation removed are sealed. Sealing is accomplished with either welded cover plates, welded caps, or blind flanges.

There are two independent trains in the PCRV water cleanup and clarification system, to allow for maintenance and repair. Each train has sufficient valves and drains to allow isolation as required.




FSV SITE
Figure 4-1

# 5.0 ADMINISTRATIVE CONTROLS

## 5.1 Responsibility

The Decommissioning Program Director shall have overall onsite responsibility for all Fort St. Vrain decommissioning activities, for both PSC and contractor personnel. The Program Director shall delegate in writing the succession to this responsibility during absences.

The Vice President responsible for nuclear activities shall have overall executive responsibility for all Fort St. Vrain decommissioning activities.

## 5.2 Organization

The decommissioning organization, functional requirements, and qualification requirements for key decommissioning personnel, for both PSC and contractor groups, shall be documented in the FSV Decommissioning Plan.

The organization responsible for quality assurance shall report to the Vice President responsible for nuclear activities on quality assurance matters, to ensure independence.

An individual qualified in radiation protection procedures shall be present at the facility at all times during physical decommissioning activities.

## 5.3 Decommissioning Safety Review Committee (DSRC)

5.3.1 The DSRC shall be comprised of the following:

Decommissioning Program Director (Chairman)
Deputy Director
Facility Support Manager (Radiation Protection Manager)
Decommissioning Engineering Manager
Operations Manager
Project Assurance Manager
Project Controls Manager
Westinghouse Project Director
Consultants may be appointed as members, in writing, by the DSRC Chairman

An alternate Chairman and alternate members, if required, shall be appointed in writing by the DSRC Chairman.

ADMINISTRATIVE CONTROLS (Continued)

5...2 The DSRC shall meet at least once per calendar quarter, or more frequently as convened by the DSRC Chairman or the Vice President responsible for nuclear activities.

5.3.3 A quorum of the DSRC shall consist of the Chairman or alternate Chairman, and a simple majority of the members, including alternates. No more than two alternate members shall participate as voting members in DSRC activities at any one time.

5.3.4 The DSRC shall be responsible for review of:

a. Administrative procedures, plans, manuals, and programs required by Specifications 5.4.1 through 5.4.4, 5.7, and permanent changes thereto, that affect RADIATION SAFETY.

b. Proposed tests and experiments that affect RADIATION SAFETY.

c. The following items, that have been evaluated to involve an unreviewed safety question as defined in 10 CFR 50.59:

1) Administrative procedures, plans, manuals, and programs required by Specifications 5.4.1 through 5.4.4, 5.7, and permanent changes thereto,

2) Proposed changes or modifications to plant systems or equipment, and

3) Proposed tests and experiments.

d. Proposed changes to the decommissioning work specifications that affect RADIATION SAFETY, and any new decommissioning work specifications that affect RADIATION SAFETY.

e. Proposed changes to the Decommissioning Technical Specifications or Facility License.

f. Investigations of violations of Decommissioning Technical Specifications, and of regulations or license requirements.

g. Reportable events as defined by 10 CFR 50.73.

h. Unplanned release of radioactive material to the environs.

ADMINISTRATIVE CONTROLS (Continued)

5.3.5 The DSRC shall:

a. Advise the Decommissioning Program Director on matters that affect RADIATION SAFETY.

b. Recommend to the Decommissioning Program Director in writing, approval or disapproval of items considered under Specifications 5.3.4.a through 5.3.4.d above.

c. Render determinations in writing with regard to whether or not each item considered under Specification 5.3.4.c constitutes an unreviewed safety question.

d. Recommend to the Decommissioning Program Director other areas of facility activities where additional oversight is prudent and/or where independent auditing is needed.

5.3.6 Audits of decommissioning activities shall be performed under the cognizance of the DSRC. These audits shall encompass:

a. A decommissioning program audit to be performed at least once per year, encompassing the following:

1) Decommissioning Technical Specifications

2) Radiation Protection Program

3) Training Program

4) Decommissioning QA Plan

5) Decommissioning Access Control Plan

6) Decommissioning Fire Protection Plan

7) Decommissioning Emergency Response Plan

b. Any other area of facility activities considered appropriate by the DSRC.

5.3.7 Records of DSRC activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each DSRC meeting and documentation of the reviews performed per Specification 5.3.4 above shall be approved and forwarded to the Vice President responsible for nuclear activities within 30 days following the meeting.

ADMINISTRATIVE CONTROLS (Continued)

b. Audit reports encompassed by Specification 5.3.7 above shall be forwarded to the Vice President responsible for nuclear activities within 30 days after completion of the audit.

5.4 Procedures and Programs

5.4.1 Written administrative procedures, plans, manuals, and/or programs shall be established, implemented, and maintained covering the activities referenced below:

a. Radiation Protection Program

b. Surveillance test activities of equipment required by these Decommissioning Technical Specifications

c. Decommissioning Access Control Plan

d. Decommissioning Emergency Response Plan

e. PROCESS CONTROL PROGRAM

f. OFFSITE DOSE CALCULATION MANUAL

g. Decommissioning Fire Protection Plan

5.4.2 Administrative procedures, plans, manuals, and/or programs of Specification 5.4.1 above, and permanent changes thereto, that affect RADIATION SAFETY, shall be reviewed by the DSRC, or a subcommittee thereof, and approved by the appropriate management prior to implementation. Procedures shall be reviewed periodically as set forth in Administrative Procedures.

Changes to the OFFSITE DOSE CALCULATION MANUAL shall be processed in accordance with Specification 5.10, and changes to the PROCESS CONTROL PROGRAM shall be processed in accordance with Specification 5.9.

5.4.3 Temporary changes to administrative procedures, plans, manuals, and/or programs of Specification 5.4.1 above may be made provided the change is documented and approved by the appropriate management prior to implementation.

ADMINISTRATIVE CONTROLS (Continued)

5.4.4 The following programs shall be established, implemented, and maintained:

a. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the OFFSITE DOSE CALCULATION MANUAL, (2) shall be implemented by procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the OFFSITE DOSE CALCULATION MANUAL,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20 limits,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20 and with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL at least every 31 days,

ADMINISTRATIVE CONTROLS (Continued)

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the EXCLUSION AREA BOUNDARY conforming to the doses associated with 10 CFR part 20,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the EXCLUSION AREA BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas beyond the EXCLUSION AREA BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

ADMINISTRATIVE CONTROLS (Continued)

b. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the OFFSITE DOSE CALCULATION MANUAL, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the EXCLUSION AREA BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

ADMINISTRATIVE CONTROLS (Continued)

## 5.5 Reporting Requirements

In addition to the applicable reporting requirements of 10 CFR, the following reports shall be submitted to the Regional Administrator of the NRC's Region IV office unless otherwise noted:

### 5.5.1 Annual Radiological Reports

Annual reports covering the activities described below, for the previous calendar year shall be submitted as follows:

a. Annual Radiation Exposure Report

The Annual Radiation Exposure Report for the previous calendar year shall be submitted to the Commission within the first calendar quarter of each calendar year in compliance with 10 CFR 20 and in accordance with the guidance contained in Regulatory Guide 1.16.

b. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the activities of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the OFFSITE DOSE CALCULATION MANUAL and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ADMINISTRATIVE CONTROLS (Continued)

5.5.2 Semiannual Radioactive Effluent Release Report

The Semiannual Radioactive Effluent Release Report covering activities during the previous 6 months shall be submitted within 90 days after January 1 and July 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The report shall also include a copy of the OFFSITE DOSE CALCULATION MANUAL, if any changes were made during the report period, as required by Specification 5.10. The material provided shall be (1) consistent with the objectives outlined in the OFFSITE DOSE CALCULATION MANUAL and PROCESS CONTROL PROGRAM, and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5.5.3 Nonroutine Reports

a. The NRC Operations Center shall be notified of emergency and nonemergency events in accordance with 10 CFR 50.72.

b. Reportable events shall be reported in accordance with 10 CFR 50.73.

5.5.4 Special Reports

Special Reports required by Specification 3.4 shall be submitted to the NRC Regional Administrator within the time period specified.

5.6 Record Retention

5.6.1 The following records shall be retained for at least three years:

a. Records and logs of principal maintenance activities, inspection, repair, and replacement of principal items of equipment related to RADIATION SAFETY.

b. Licensee Event Reports (LERs).

c. Records of surveillance activities, inspections, and calibrations required by the Decommissioning Technical Specifications.

ADMINISTRATIVE CONTROLS (Continued)

d. Records of changes made to procedures related to RADIATION SAFETY.

e. Records of radioactive shipments.

f. Records of sealed source leak tests and results.

5.6.2 The following records shall be retained for the duration of the Facility License:

a. Dismantlement records for systems and equipment related to RADIATION SAFETY.

b. Records of facility radiation and contamination surveys, including final site release records.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of training and qualification for current members of the decommissioning staff.

f. Records of activities required by the Decommissioning QA Plan.

g. Records of reviews performed pursuant to 10 CFR 50.59.

h. Records of meetings of the DSRC.

i. Records and logs pertaining to the Radiological Environmental Monitoring Program.

j. Records of changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

5.7 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained, and adhered to for all activities involving personnel radiation exposure.

ADMINISTRATIVE CONTROLS (Cont.,ued)

5.8 High Radiation Area

5.8.1 Pursuant to 10 CFR 20, in lieu of the "control device" or "alarm signal", each high radiation area, as defined in 10 CFR Part 20, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics personnel) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area, or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them, or

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physics staff in the RWP.

ADMINISTRATIVE CONTROLS (Continued)

5.8.2 In addition to the requirements of 5.8.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in) from the radiation source or from any surface which the radiation penetrates shall be provided with locked enclosures to prevent unauthorized entry, and the keys shall be maintained under the administrative control of health physics supervision. Enclosures shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in the area. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

For individual areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device whenever the dose rate in the area exceeds or will shortly exceed 1000 mR/hr.

5.9 PROCESS CONTROL PROGRAM (PCP)

Permanent changes to the PROCESS CONTROL PROGRAM:

a. Shall be documented and records of reviews performed shall be retained as part of the DSRC meeting records, as required by Specification 5.6.2. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

   2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the DSRC in accordance with Specification 5.3.6.

ADMINISTRATIVE CONTROLS (Continued)

5.10 OFFSITE DOSE CALCULATION MANUAL

Changes to the OFFSITE DOSE CALCULATION MANUAL:

a. Shall be documented and records of reviews performed shall be retained as part of the DSRC meeting records, as required by Specification 5.6.2. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20, 10 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by DSRC in accordance with Specification 5.3.6.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire OFFSITE DOSE CALCULATION MANUAL as a part of or concurrent with the Semiannual Radioactive Effluent Release Report for the period of the report in which any change to the OFFSITE DOSE CALCULATION MANUAL was made.

5.11 Natural Gas Restriction

As indicated in Specification 1.0, FSV is being converted to utilize a gas-fired boiler. The natural gas line supplying this boiler, or any other new natural gas source, shall not be introduced within 0.5 miles of the location where ACTIVATED GRAPHITE BLOCKS are stored, for any purpose, without prior NRC approval. PSC shall submit an analysis of any proposed new natural gas source demonstrating that the new source will not present an unacceptable hazard to the ACTIVATED GRAPHITE BLOCKS or to the equipment or systems needed to protect the ACTIVATED GRAPHITE BLOCKS.

ATTACHMENT 3

TO P-92115

JUSTIFICATION OF CHANGES

FROM PREVIOUSLY SUBMITTED

FACILITY LICENSE AND

DECOMMISSIONING TECHNICAL SPECIFICATIONS

## JUSTIFICATION OF CHANGES FROM PREVIOUSLY SUBMITTED FACILITY LICENSE AND DECOMMISSIONING TECHNICAL SPECIFICATIONS

1. Facility License Section I.A(4) was revised to allow receipt of byproduct and special nuclear materials that were produced by the previous operation of the facility or during decommissioning activities.

   Justification:

   PSC's initial proposed Facility License authorized the licensee to possess but not separate byproduct and special nuclear material, consistent with the current Facility Operating License. The added authorization to **receive** materials produced during previous operation or during decommissioning activities is needed to provide flexibility for waste processing during decommissioning. For example, radioactive waste disposal volumes can be minimized by volume reduction and decontamination. The proposed license revision would allow PSC to ship radioactive materials offsite to a commercial volume reduction or decontamination firm, and then receive it back for staging and/or packaging for shipment to its ultimate disposal site.

   The proposed license revision limits PSC to receiving only that material produced by previous operation of the plant or during decommissioning activities (e.g., decontamination wipes). This provision does not create the potential for FSV to become a temporary storage facility for other facilities' wastes.

2. Management titles in Section 5 were revised to reflect PSC's revised management organization. This includes the following changes:

   All references to the Vice President, Nuclear Operations have been replaced with the Vice President responsible for nuclear activities. This designates the executive level position with responsibility for all activities associated with Fort St. Vrain.

   All references to the Program Manager for Decommissioning have been replaced with the Decommissioning Program Director.

   The Deputy Director and the Project Controls Manager have been added to the Decommissioning Safety Review Committee.

Justification:

The re-organization reflected in the above titles is illustrated on the attached organization chart, provided for information. This organization does not represent a significant change from that described in Section 2.4 of the Proposed Decommissioning Plan. The duties of the Program Manager have been divided between the Program Director and the Deputy Director, so the Deputy Director was added to the Decommissioning Safety Review Committee (DSRC). The inclusion of the Project Controls Manager on the DSRC ensures representation of all decommissioning departments.

Designation of the Vice President responsible for nuclear activities designates an executive level position, and provides flexibility by allowing certain duties to be performed by either the Vice President or the appropriate Senior Vice President.

3. The DSRC quorum requirements in Section 5.3.3 were revised from three members to a simple majority of the members.

Justification:

This revision provides for a more accurate determination of the quorum requirements. Seven members are listed in Section 5.3.1, in addition to the Chairperson, and various consultants may also be appointed as members. Therefore, the total number of members may change and three members may not adequately define a quorum.

4. The time allowed to submit the Semiannual Radioactive Effluent Release Report in Section 5.5.2 was revised from 60 days to 90 days.

Justification:

Submittal of the Semiannual Radioactive Effluent Release Report within 60 days of the cutoff date is a provision of the standard technical specifications, but it is not actually required by regulation. PSC considers that this time requirement will be an unnecessarily onerous burden during decommissioning, and proposes that 90 days be allowed to assemble all the pertinent data and submit the report.

PSC's staffing levels during decommissioning will be reduced to a level that will permit oversight of our decommissioning contractor. Historically, the preparation of the Semiannual Radioactive Effluent Release Reports has consumed considerable resources, and PSC is concerned that our reduced staffing levels during decommissioning may not support a 60 day submittal. PSC recognizes the value of the information contained in this report, and requests that 90 days be allowed after the cutoff dates to provide this information. PSC considers this to be a reasonable request considering the limited duration of decommissioning activities at FSV.

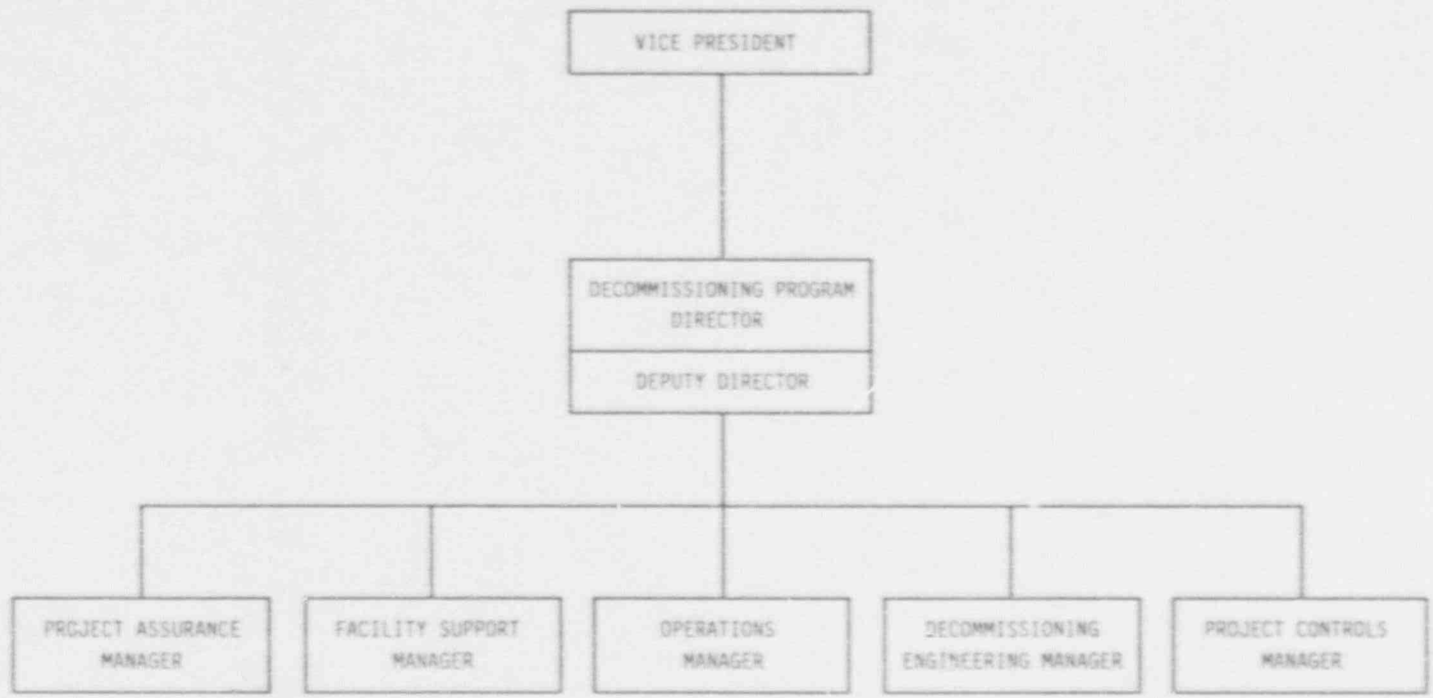
5. In the requirements for high radiation area controls, the criteria that define a high radiation area were deleted from Section 5.8.1, and Section 5.8.2 was revised to add clarification that a flashing light warning device shall be used "whenever the dose rate in the area exceeds or will shortly exceed 1000 mR/h."

Justification:

Some of the FSV decommissioning activities may be completed before the implementation date of the revised 10 CFR 20. The deletion of wording that is unique to a specific version of 10 CFR 20 will provide necessary flexibility to utilize whichever version of the regulation is in effect.

The phrase added to Section 5.8.2 regarding the flashing light warning device requirements was added for clarification, and was obtained from draft regulatory guide DG-8006, Control of Access to High and Very High Radiation Areas in Nuclear Power Plants.

FSV DECOMMISSIONING
ORGANIZATION



ATTACHMENT 4

TO P-92115

NO SIGNIFICANT HAZARDS

CONSIDERATION ANALYSIS

DECOMMISSIONING OF THE FORT ST. VRAIN
NUCLEAR GENERATING STATION

NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

## INTRODUCTION

Pursuant to 10 CFR 50.92, each application for amendment to an operating 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 [10CFR50.92(c)]. A proposed amendment to an operating 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.

The amendment, as defined below, describing the replacing of the existing Technical Specifications, in their entirety, with the Decommissioning Technical Specifications has been reviewed and deemed not to involve a significant hazards consideration. The basis for this determination follows.

## BACKGROUND

Public Service Company of Colorado (PSC) is proposing to decommission the Fort St. Vrain Nuclear Generating Station. Pursuant to 10 CFR 50.82, PSC has prepared and submitted a Proposed Decommissioning Plan (PDP) to the NRC for review and approval. The decommissioning of nuclear facilities is a regulated process whereby radioactive material is removed from the plant site, the site is decontaminated to established limits, and NRC licenses are terminated.

The current requirements with regard to occupational or public doses or effluents to the environment continue to apply throughout the decommissioning period until the license is terminated by the Commission. The decommissioning planning requirements are considered appropriate means of assuring that the decommissioning will be carried out in accordance with 10CFR Part 20, and specifically that the doses will be kept as low as reasonably achievable (ALARA).

PSC has selected the DECON option for decommissioning Fort St. Vrain. PSC is proposing the immediate dismantlement and decommissioning of Fort St. Vrain to release all site areas for unrestricted use. To accomplish this, the following activities will be undertaken:

1. Removal of the Prestressed Concrete Reactor Vessel (PCRV) internal radioactive components remaining after the defueling of the reactor.

2. Decontaminate and/or dismantle those portions of the PCRV structure and radioactive balance-of-plant systems which exceed limits for unrestricted release of residual radioactive materials.

3. Ship all radioactive waste offsite for disposal.

4. Perform a final site radiation survey to confirm that all site areas can be released for unrestricted use.

5. Terminate the 10CFR50 operating license.

The major dismantling and decontamination activities that will be performed during decommissioning are described in detail in Section 2.3 of the PDP and are summarized below:

Decontamination and Dismantlement of the Prestressed Concrete Reactor Vessel

The major decommissioning task is the dismantlement and decontamination of the radioactive portions of the PCRV. Initial dismantlement of the PCRV will include removal of selected PCRV internal components and removal of portions of the steam generators. Simultaneously, the non-contaminated portion of the steam generators will be removed from the lower portion of the PCRV to provide access for detachment of the contaminated steam generator upper assemblies. After the steam generator lower assemblies are removed from the bottom of the PCRV, the PCRV bottom head and side wall penetrations will be sealed, a water cleanup and clarification system will be connected, and the PCRV will be flooded. Flooding of the PCRV will provide shielding for the workers associated with PCRV dismantlement activities.

To allow removal of internal core components, a plug of concrete will be removed from the top head of the PCRV. This opening will be used to remove any defueling elements and hexagonal reflector blocks that had not previously been removed by use of the Fuel Handling Machine. Other internal core components including the large side reflector blocks, side spacer blocks, core support blocks, and core support posts will also be removed through this top head opening. After the core internals have been removed, the core barrel will be removed by cutting it into pieces sized for disposal in radwaste containers. Following removal of the core barrel, the PCRV water level will be lowered and the core support floor (CSF) insulation will be removed. The CSF will then be lifted to the top of the PCRV by means of a system of cables and hydraulic jacks, where it will then be segmented into pieces and removed using the Reactor Building crane. Once the CSF is removed, the helium circulator diffusers and steam generator upper modules will be removed. The remaining radioactive components, which include the activated "beltline concrete" of the PCRV around the reactor core region, the PCRV liner, liner insulation and insulation cover plates, and the PCRV lower floor with its supports will also be removed.

Decontamination and Dismantlement of Contaminated Balance of Plant (BOP) Systems-

The decontamination and dismantlement of contaminated or potentially contaminated balance of plant systems will be performed by either decontamination in place, removal and decontamination, or removal and disposal as radioactive waste. In general, contaminated or potentially contaminated piping, components, structures, walls and ductwork will be surveyed to determine acceptability for unrestricted release or to determine the cleanup required for release.

EVALUATIONS

Accidents that could result in radiation exposure at the site boundary during the dismantling and disposal activities have been postulated and analyzed. However, since the reactor will be defueled prior to the commencement of decommissioning operations and all fuel will be removed from the Reactor Building, the risk of accidents resulting in a radiological release during decommissioning activities is considerably less than during plant operation. Furthermore, the accident analyses have determined that the radiation doses to the general public from postulated accidents would be a small fraction of the Protective Action Guide levels recommended by the U.S. Environmental Protection Agency (Reference 1).

Section 3.4 of the Proposed Decommissioning Plan describes the analyses for the following postulated accidents:

Dropping of contaminated concrete rubble.
Conversion construction near PCRV dismantlement.
Heavy load drop.
Fire.
Loss of PCRV shielding water.
Loss of power.
Natural disasters.
Steam generator primary module drop (Analysis was provided in PSC's February 18, 1992 response to RAI questions 9, 12, and 14, P-92064, and will be incorporated into PDP Section 3.4 at its next revision).

The components with the highest potential radiation release were considered in the accident analyses. Therefore, accidents that were analyzed bound the radiological consequences from other postulated accident scenarios. In evaluating the postulated accidents, conservative assumptions were made when data or knowledge to support more realistic analyses were lacking. Conservatism in this context means that the radiological consequences from the postulated accidents will be overestimated rather than underestimated.

A capsule summary of the accident scenarios is given in Table 3.4-1 of the PDP. A distance of 100 meters was used as the exclusion area boundary (EAB), the distance from the Reactor Building walls to the nearest fenced site boundary. A worst case atmospheric dispersion factor of 3.53 E-2 sec/m$^3$ has been calculated and is used in the accident analyses, with the exception of the Tornado accident, which utilizes an atmospheric dispersion factor of 4.59 E-4 sec/m$^3$. The atmospheric dispersion factor of 4.59 E-4 sec/m$^3$ represents an annual average dispersion factor for Fort St. Vrain, and is believed to be conservative in the event of a tornado.

These atmospheric dispersion factors were calculated using the guidelines presented in Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, and conservatively assumed a wind speed of 1 mph (1 mph for the worst case and 5 mph for annual average). The analyses also assumed that all releases to the environment were ground level releases. The major exposure pathway was assumed to be air inhalation, with the lung representing the critical organ.

The doses to an offsite individual from the postulated accident scenarios are presented in Table 3.4-2 of the PDP. An inspection of this table reveals that the limiting accident is a fire with a whole body dose of 121 millirem and a 215 millirem dose to the organ (lung). The results of the accident analyses indicate that the radiation exposures to the general public will be very low. In all cases, the radiological consequences from the postulated decommissioning accident scenarios are well within the 25 rem whole body dose and 300 rem to any specific organ guidelines established by 10CFR100. The radiological consequences are also 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.

For comparison with previously analyzed accidents, Section 14.11 of the Fort St. Vrain FSAR (Reference 2) provides a discussion of Design Basis Accident Number 2 (Rapid Depressurization/Blowdown). As stated in the FSAR, this accident postulates a hypothetical sudden failure of both closures in a PCRV penetration resulting in the release of the fission product inventory circulating within the primary coolant and a fraction of the plateout activity. For this accident scenario it was conservatively assumed that the coolant escapes directly from the building into the atmosphere at ground level without any credit for holdup or filtration by the ventilation system. The resultant doses at the EAB (590 meter boundary) from this analysis are 2.5 rem whole body gamma, 17.4 rem thyroid (the highest inhalation organ dose) and 4.8 rem bone.

The analysis determined that the radiological consequences of the rapid depressurization would be within the guidelines of 25 rem whole body and 300 rem to any specific organ prescribed by 10CFR100. As presented in the PDP, the worst case decommissioning accident is a postulated fire occurring at ground level immediately outside of the Reactor Building truck loading bay resulting in EAB (100 meter boundary) doses of 121 millirem to the whole body and 215 millirem to the lung (the highest inhalation organ dose). No credit was taken for holdup or filtration by the ventilation system. The radiological consequences from this postulated accident are significantly less than those from the Design Basis Accident Number 2. These accident scenarios are judged to be comparable and the same type of accidents since both involve a sudden release of radioactivity into the atmosphere. Due to the low consequences (a small fraction of EPA Protective Action Guidelines) of postulated decommissioning accidents, it can be concluded that the Fort St. Vrain decommissioning activities do not pose an undue risk to the health and safety of the general public.

## CONCLUSIONS

Since the reactor will be defueled prior to the commencement of decommissioning operations and all irradiated fuel will be removed from the Reactor Building, there will be no need for shutdown/cooldown systems such as decay heat removal nor is there need for safety systems pertaining to reactivity control. All plant systems which will be relied upon during decommissioning are described in the PDP and are governed by the Decommissioning Technical Specifications.

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

1. Superseding the existing Technical Specifications by the Decommissioning Technical Specifications does not increase the probability or consequences of an accident previously evaluated in the FSAR. Ceasing plant operations and removing all irradiated fuel from the Reactor Building eliminates the probability of power operations and refueling accidents that are evaluated in the Fort St. Vrain FSAR. The probability of occurrence of the accidents analyzed in the PDP are generally quite low. An accident which could reasonably be expected to occur over the course of decommissioning is the loss of power. However, the probability of such an occurrence is not significantly different from the loss of outside electric power events analyzed in FSAR Section 10.3. Heavy load drops of highly activated or contaminated components are not anticipated during decommissioning, and are not considered to be significantly more probable than the load drops assessed in FSAR Section 9.2.11. Should such load drops occur, Decommissioning Technical Specifications 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.

   The probability of a fire which could result in the release of significant quantities of activity, such as analyzed in the PDP, is considered to be extremely low. A rupture which would result in PCRV shielding water flowing into the Reactor Building is not expected to occur over the course of decommissioning, although the consequences of such an event are negligible since the water inventory is contained by the Reactor Building and only small amounts of tritium are released due to evaporation. Decommissioning Technical Specification 3.4 assures tritium concentrations will not exceed those assumed in the PDP accident analyses. The probability of natural disasters, such as earthquakes or tornadoes, is unchanged.

   The radiological effluent disposal system specifications, Section 8 of the existing Technical Specifications are not included in the Decommissioning Technical Specifications. However, Specifications 5.9, "Process Control

Program" and 5.10, "Offsite Dose Calculation and Radiological Environmental Monitoring Program Manuals", in conjunction with Decommissioning Technical Specifications 2.9 and 2.11, require controls associated with radiological effluents to be maintained to assure compliance with 10CFR20 requirements. Removal of detailed requirements for radiological effluents from the Technical Specifications, and incorporation of these requirements in programs referenced by the Technical Specifications, was approved by the NRC in NRC Generic Letter 89-01, dated January 31, 1989. Since process controls on radiological effluents will remain, this activity does not increase the probability of uncontrolled effluent releases which could exceed 10CFR20 limits.

Based on the above, the probability of occurrence of accidents and malfunctions which could result in release of radioactivity is not significantly different from similar accidents and malfunctions evaluated in the FSAR.

With respect to the consequences of accident analyses, all fissionable material will be removed from the Reactor Building prior to commencing decommissioning activities. As such, accidents involving fissionable material are no longer credible. Therefore, the risk of accidents resulting in a radiological release during decommissioning activities is considerably less than during plant operation. Detailed analyses of postulated decommissioning accidents have been performed for the Proposed Decommissioning Plan. These analyses have determined that the worst case accident would result in a whole body dose of 121 millirem and a 215 millirem dose to the organ (lung) at the EAB (100 meter boundary). The results of the accident analyses indicate that the radiation exposures to the general public will be very low. In all cases, the radiological consequences from the postulated decommissioning accident scenarios are well within the 25 rem whole body dose and 300 rem to any specific organ guidelines established by 10CFR 100 and also lower than accidents involving radioactive releases previously evaluated in the FSAR. Therefore, the decommissioning accident scenarios are bounded by the current design basis accident analyses, such that no increase in radiological consequences will result.

2. The issuance of the Decommissioning Technical Specifications to supersede the existing Technical Specifications does not create the possibility of different types of accidents or malfunctions than those evaluated previously in the FSAR. Issuance of the Decommissioning Technical Specifications will provide assurance that structures, systems and equipment relied on to prevent or mitigate the consequences of accidents postulated to occur during decommissioning will perform their intended safety function. The majority of the existing Technical Specifications apply to components and systems relied upon to mitigate the consequences of accidents analyzed in the FSAR, such

as reactivity excursions, loss of cooling accidents, etc., whose occurrence are no longer credible during decommissioning conditions, with all fuel removed. Deletion of these obsolete specifications, and inclusion of the specifications required to mitigate the consequences of postulated decommissioning accidents, is warranted. The Proposed Decommissioning Technical Specifications do not place plant systems in configurations conducive to the occurrence of accidents or malfunctions not previously evaluated. The integrity of the Reactor Building, in conjuction with operation of the ventilation exhaust system, limits the off-site doses under normal and abnormal conditions during decommissioning activities. The controls will assure the Reactor Building confinement and the Reactor Building ventilation exhaust system are available to mitigate an activity release whenever decommissioning operations give rise to the possibility of a release which could exceed a small fraction of the EPA Protective Action Guidelines. Controls also provide assurance that tritium activity concentrations in the PCRV shield water will not exceed those postulated in the decommissioning accident analyses. These Section 3 General Requirements, as well as additional Administrative Controls delineated in Section 5 of the PDP, serve to promote radiation safety during decommissioning, and do not contribute to the possibility of different types of accidents or malfunctions.

Many of the categories of accidents, or accident types, evaluated in the FSAR are not possible during decommissioning. Since all irradiated fuel will be removed from the Reactor Building, accidents related to reactivity excursions, power-to-flow mismatch, and loss of decay heat removal are no longer a concern. An accident type evaluated in the FSAR which is applicable during decommissioning operations is a breach or malfunction of containment permitting release of radioactivity. Failures of the reactor coolant pressure boundary resulting in radioactive releases are evaluated in FSAR Sections 10.2.3.4, "Radiation Monitoring of Hot Reheat Piping," 14.7, "Primary Coolant Leakage", 14.8, "Maximum Credible Accident", and 14.11, "Design Basis Accident No. 2- Rapid Depressurization/Blowdown". Malfunctions of auxiliary systems resulting in leakage of radioactivity are evaluated in FSAR Section 14.6, which includes failures of the helium purification system, accidental release of the contents of a gas waste surge tank, fuel handling and fuel storage accidents and drop of a fully loaded spent fuel shipping cask.

While specific decommissioning accidents differ from these containment failure accidents, in that fuel is not present during decommissioning and the PCRV is open to Reactor Building atmosphere and flooded with water, they generally fall into the same accident category, release of radioactivity due to containment failure. Design Basis Accident No. 2 - Rapid Depressurization/Blowdown, is the most severe of this type of accident analyzed in the FSAR. FSAR Section 14.11 identifies the offsite dose consequences of this accident to an individual at the current 590 meter EAB as 2.5 rem whole body gamma, 17.4 rem thyroid and 4.8 rem bone. The consequences of this accident are much more severe than those of the postulated decommissioning accidents analyzed in the PDP. In that the possible range of accidents and malfunctions which could occur over the course of decommissioning are the same type of accident as previously evaluated in the FSAR under the category of primary coolant leaks and auxiliary system leaks, bounded by Design Basis Accident No. 2, no new types of accidents or malfunctions are being created.

3. The postulated accident analyses have verified that decommissioning activities will be maintained within the bounds of safe, analyzed conditions as defined in the Proposed Decommissioning Plan and governed by the Decommissioning Technical Specifications. The evaluation has taken into account the applicable Decommissioning Technical Specifications and has bounded the conditions under which the specifications permit decommissioning. The results presented in the FSAR are bounding. As such, the margin of safety, as defined in the bases to the Decommissioning Technical Specifications, is not reduced by replacing the current Technical Specifications, in their entirety, with the Decommissioning Technical Specifications.

Based upon the preceding evaluation, it has been determined that the proposed change to replace the current Technical Specifications with the Decommissioning Technical Specifications at the Fort St. Vrain Nuclear Generating Station does 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. Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, EPA-520/1-75-001-A, January 1990, U.S. Environmental Protection Agency.

2. Fort St. Vrain Updated Final Safety Analysis Report, Revision 9, Public Service Company of Colorado.