ATTACHMENT 2

TO P-92283

PROPOSED CHANGES TO DECOMMISSIONING

TECHNICAL SPECIFICATIONS

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 conta n 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 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.

3.1 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 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.

3.2 REACTOR BUILDING VENTILATION EXHAUST SYSTEM

LC 3.2 The Reactor Building ventilation exhaust system shall be OPERABLE w

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 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

ADMINISTRATIVE CONTROLS (Continued)

5.3.2 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 nuclear safety.

b. Proposed tests and experiments that affect nuclear safety.

c. The following items which 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 nuclear safety, and any new decommissioning work specifications that affect nuclear 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 nuclear 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 nuclear 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)

5.5.2 Annual Radioactive Effluent Release Report

The Annual Radioactive Effluent Release Report covering activities during the previous 12 months shall be submitted within 90 days after January 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, inspections, repair, and replacement of principal items of equipment related to nuclear 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 nuclear 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 nuclear 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 (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 barricaded, 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.

ATTACHMENT 3

TO P-92283

SAFETY ANALYSIS
AND
JUSTIFICATION OF CHANGES

SAFETY ANALYSIS
AND
JUSTIFICATION OF CHANGES

This attachment describes each change, provides justification for the change, and concludes with an overall safety analysis.

1. The defined term "RADIATION SAFETY" was deleted and replaced with the undefined term "nuclear safety". This change was made throughout Section 5, Administrative Controls, and the accompanying Definition 2.12 was deleted.

   Justification:

   The term "Radiation Safety" was used to distinguish decommissioning activities that are subject to various administrative controls such as safety committee review and record retention. Revising this term to "nuclear safety" is consistent with administrative controls established in the Standard Technical Specifications, in the historical FSV operating Technical Specifications, and in Technical Specifications for other plants undergoing decommissioning. This change ensures that the same level of safety committee review is provided for activities at Fort St. Vrain as is required at operating nuclear plants or other decommissioning plants.

   The term "nuclear safety" is typically not defined in a facility's Technical Specifications and it is normally left up to individual utilities to define exactly what it involves for their specific plant. For operating nuclear facilities, "nuclear safety" generally involves reactivity control, reactor cooling, and containment considerations. For Fort St. Vrain, which is undergoing decommissioning and has no nuclear fuel on-site, reactivity control and reactor cooling are not safety concerns. In this case, the concept of "nuclear safety" would apply mainly to activities which have the potential for release of radioactivity.

2. The Applicability sections for both LC 3.1 and LC 3.2 were revised to state "Whenever ACTIVATED GRAPHITE BLOCKS have been removed from the PCRV..." instead of "Whenever ACTIVATED GRAPHITE BLOCKS have been removed from the PCRV shielding water..."

Justification:

These specifications provide requirements to ensure Reactor Building confinement integrity, subatmospheric conditions, and operability of the Reactor Building ventilation exhaust fans and filters, to limit off-site doses under normal and abnormal conditions during decommissioning activities. These Reactor Building conditions are relied upon in decommissioning accident analyses (See Section 3.4 of the Proposed Decommissioning Plan) involving handling of activated graphite blocks.

The Proposed Decommissioning Plan (Section 2.3.3.8) discusses removal of hexagonal graphite blocks from the Prestressed Concrete Reactor Vessel (PCRV) by means of either the Fuel Handling Machine under dry conditions, or by use of special tooling after the PCRV is flooded with shielding water. The Applicability sections of LC 3.1 and LC 3.2 originally considered that the PCRV would be filled with shielding water before any activated graphite blocks were removed. However, the hexagonal graphite blocks are being removed with the Fuel Handling Machine to take advantage of its availability, good performance, and shielding features. Thus, some activated graphite blocks (e.g., hexagonal reflector blocks) are being removed from the PCRV before any shielding water has been added, and Reactor Building confinement integrity and subatmospheric conditions should continue to be required during these activities. By revising the specification Applicability statements to apply whenever activated graphite blocks are removed from the PCRV, appropriate controls will be provided.

3. Administrative Control paragraph 5.3.4.c was revised to state that the DSRC shall review "The following items which involve an unreviewed safety question...", instead of "The following items, that have been evaluated to involve an unreviewed safety question..."

Justification:

This change was made for clarification. In addition to other items specified for DSRC review, paragraph 5.3.4.c has been revised to clarify that the DSRC shall review those procedures, changes, modifications, tests, and experiments that involve an unreviewed safety question. This is consistent with the current FSV Technical Specification requirements for the Nuclear Facility Safety Committee (NFSC) and with Standard Technical Specification requirements for the Company

Nuclear Review and Audit Group (CNRAG) functions (See Westinghouse STS, NUREG-0452, Revision 5, Section 6.5.2.7).

The Administrative Controls of the FSV Decommissioning Technical Specifications provide for a single Decommissioning Safety Review Committee (DSRC), that combines the functions of the current Plant Operating Review Committee (PORC) and the NFSC. The DSRC reviews items that affect nuclear safety, similar to the current PORC functions, and it also reviews items that involve an unreviewed safety question, similar to the current NFSC functions. This proposed revision retains those functions for the DSRC.

4. The Semiannual Radioactive Effluent Release Report described in Administrative Controls section 5.5.2 was revised to an Annual Radioactive Effluent Release Report.

Justification:

This change is consistent with a revision to 10 CFR 50.36a, dated August 19, 1992, published in the Federal Register on August 31, 1992 (57 FR 39358). As noted in the supplementary information for this rule (57 FR 39355), this change was made specifically to reduce the requirements for reports concerning the quantity of principal nuclides released to unrestricted areas in liquid and gaseous effluents from semiannually to annually.

5. The High Radiation Area provisions in Administrative Controls section 5.8.2 were revised to require that high radiation areas (>1000 mr/hr) be "barricaded" instead of "roped off."

Justification:

This change is made for clarification, and is consistent with the requirements in the Standard Technical Specifications (See Westinghouse STS, NUREG-0452, Revision 5, Section 6.12.2). The previous provision to "rope off" high radiation areas reflects PSC's normal practices, but it is overly prescriptive. The provision to "barricade" high radiation areas is consistent with controls provided for operating nuclear plants and it allows the use of any barrier that obstructs passage, including ropes.

Safety Analysis:

The above changes do not involve an unreviewed safety question as defined in 10 CFR 50.59, as follows:

a) The above changes do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report (either the current FSAR or the Proposed Decommissioning Plan).

The changes described above are largely administrative in nature. These changes ensure safety committee review consistent with operating plants and other plants undergoing decommissioning. Also, the Applicability of specifications for Reactor Building confinement integrity and ventilation exhaust fan and filter operability is changed to ensure that the requirements are applied during activated graphite block handling activities. This ensures that potential off-site doses would be limited for any accident involving activated graphite blocks within the Reactor Building. These controls are consistent with the assumptions in the Proposed Decommissioning Plan accident analysis, Section 3.4.

b) The above changes do not create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report (either the current FSAR or the Proposed Decommissioning Plan).

No new equipment operating modes or conditions are created by the above changes and no reduction in administrative controls is involved. The changes described above could not create any new accident or malfunction.

c) The above changes do not reduce the margin of safety as defined in the basis for any technical specification.

The above changes do not affect any technical specification bases discussions. In that these changes are largely administrative in nature, no margin of safety is affected. The reduction in frequency of the Radioactive Effluent Release Report from semiannual to annual is based on a Federal Rule and is therefore acceptable. Also, the change in requirements for high radiation area barriers from "roped off" to "barricaded" is more generic and ensures adequate worker protection consistent with Standard Technical Specification requirements.

ATTACHMENT 4

TO P-92283

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 usin, 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 PC'.V 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 ... 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.