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Fort St. Vrain
Unit No. 1
P-88205

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

Attention: Mr. Jose A. Calvo
Director, Project Directorate IV

Docket No. 50-267

SUBJECT: Technical Specification Upgrade Program (TSUP), Justification of Changes to Current Specifications

REFERENCES: 1) PSC letter, Brey to Calvo, dated 5/27/88 (P-88184)

2) PSC letter, Brey to Calvo, dated 3/8/88 (P-88082)

Dear Mr. Calvo:

This letter provides information to support the NRC review of the revised final draft of the upgraded Fort St. Vrain (FSV) Technical Specifications, submitted by Public Service Company of Colorado (PSC) in Reference 1.

Attachment 1 provides discussions for each NRC comment that had previously been determined to require written justification. These comments address TSUP positions that are less restrictive than the current FSV Technical Specifications. In previous discussions with the NRC, the TSUP positions were deemed conditionally acceptable pending futher justification, and were designated as category "C" comments.

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Attachment 2 includes revised drafts for three TSUP specifications that PSC has changed subsequent to Reference 1. This attachment also discusses the reasons for the revisions.

Attachment 3 provides revised discussions regarding the impact that recent Amendment 51 to the FSV Technical Specifications has on the TSUP specifications. This amendment provided specific Inservice Inspection surveillance requirements that, for certain equipment, PSC had previously proposed to delete from TSUP (Reference 2). On further review, however, the surveillances in question were reinstated and were included in the Revised Final Drafts of Reference 1.

If you have any questions regarding this information, please contact Mr. M. H. Holmes at (303) 480-6960.

Very truly yours,

Lawrence Brey

H. L. Brey, Manager
Nuclear Licensing and
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HLB/SWC/lmb

Attachments

cc: Regional Administrator, Region IV
ATTN: Mr. T. F. Westerman, Chief
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Mr. R. E. Farrell
Senior Resident Inspector
Fort St. Vrain

JUSTIFICATION FOR
CATEGORY "C" COMMENTS

This attachment provides justifications for each NRC comment regarding the TSUP that had been categorized as a "C" comment during previous discussions. These are comments where general agreement was reached during discussions between PSC and the NRC, but written justification is required because the TSUP position involves a less restrictive change to the current FSV Technical Specifications.

The attached discussions are consistent with the style, content, and level of detail provided in the sample discussions in Attachment 5 to P-87063. Each comment is identified relative to its source document. For each TSUP position that involves a deletion or reduction in the requirements of the current technical specification, PSC has provided a significant hazards consideration review per 10CFR50.92.

Three of the attached discussions were previously submitted in Attachment 2 to P-85098, dated 4-1-85. These address the deletion of technical specifications for the Breathing Air System, the core differential pressure instrumentation, and the Rise-to-Power testing. These discussions are repeated here for completeness.

Comment SL 2.2.1-5 was previously designated as a "C" comment in NRC memorandum, Heitner to Calvo, dated 10/1/87 (G-87348). PSC considers that this was intended to refer to comment SL 2.1.1-5. Comment SL 2.1.1-5 is the "C" comment that was discussed in the meeting documented by G-87348. Comment SL 2.2.1-5 addresses the addition of the detector decalibration curves to the LSSS Section of TSUP, which is clearly a conservative change that was agreed to with no further justification required. Comment SL 2.2.1-5 has not been addressed herein.

NRC Comment: SL 2.1.1-5 (G-86285)

Current Technical Specification SL 3.1 has been rewritten as a safety limit (SL 2.1.1) and as a limiting condition for operation (LCO 3.2.6) to clarify the original intent and implement the requirements in their appropriate forms.

The current SL 3.1 is intended to assure the integrity of the fuel particles as a fission product barrier by observance of a limit on the combination of reactor core power-to-flow ratio and the total integrated operating time at a particular power-to-flow ratio (i.e., Fig. 3.1-1). For the purpose of determining the operating time at a particular power-to-flow ratio, only transients exceeding a screening criteria (i.e., Fig. 3.1-2) are considered.

These same limits have been retained in the TSUP; they have been clarified, consistent with the goals of the TSUP. The actual safety limit identified in current SL 3.1 has not been changed.

This approach is consistent with the practice of the STS wherein LCO requirements would have to be exceeded before related safety limit requirements would be reached.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. The TSUP includes all of the limitations, actions, and surveillances of the current technical specifications. Although in a different format, the upgraded technical specifications provide at least the same level of assurance of fuel particle integrity as the current specifications. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

2. Likewise, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. Although in a different format, there is no change from current requirements in effect.

3. This change does not involve a significant reduction in a margin of safety. Incorporating all the limitations, actions, and surveillances of the existing safety limit assures that the margin of safety is maintained.

NRC Comment: LCO 3.1.1-9 (G-86285)

When reactor pressure is above 100 psia, TSUP specification 3/4.1.1 requires a helium purge flow to each CRD penetration for partially or fully withdrawn control rod pairs. A minimum purge flow has not been specified, as previously requested by NRC, for the following reasons.

The purpose of the helium purge flow is to limit the upward flow rate of hot, contaminated helium from the core to the control rod drive mechanism. Maintaining any purge flow, no matter how small, helps to control CRDM temperature and minimizes CRDM contamination simply by preventing the upward flow of reactor coolant. It is not clear that quantification of a minimum flow requirement would be meaningful.

The need for helium purge flow is part of the larger issue of long term control rod drive operability currently under investigation by PSC. Future resolution of outstanding commitments to resolve the integrated CRD operability issues will address purge flow requirements as appropriate. At this point, however, PSC does not consider a minimum purge flow specification appropriate.

This is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. The operability of control rod drive mechanisms is assured by partial and full scram tests, by monitoring CRD motor temperature, by verifying the presence of purge flow, by verifying the absence of moisture in the purge flow, and by preventive maintenance. These measures are specifically intended to assure that there is no significant increase in the probability of an accident previously evaluated. In the event of an accident involving the control rod drive mechanisms, purge flow has no effect on the consequences of accidents previously evaluated.

2. The absence of a specific minimum limit on purge flow cannot create the possibility of a new or different kind of accident from any accident previously evaluated. PSC considers that control rod drive accidents have been fully evaluated and CRD purge flow does not significantly affect any other plant system, component or structure.

3. This does not involve a significant reduction in a margin of safety. The pertinent margin of safety is provided by control rod operability which is already assured in a comprehensive manner by the technical specifications, including a requirement that purge flow must be maintained.

NRC Comment: LCO 3.6.5.2-9 (G-86285)

The May 25, 1988 Draft of TSUP Specification SR 4.6.5.2 required testing of the charcoal adsorber material in the reactor building ventilation filters after painting in areas that communicate with the ventilation system, unless low solvent paints are used. This exception is not consistent with the STS guidance, but was proposed to permit controlled painting activities in the FSV reactor building, which is normally accessible during operations.

Upon further consideration, PSC has revised the SR to delete the exception for low solvent paints and has defined the administrative controls for painting activities in the BASIS. The reactor plant exhaust system testing requirements are invoked after significant painting, when the quantity of paint used in the reactor building during the normal 6 month surveillance interval exceeds 5 gallons. The justification for this follows.

The FSV reactor building is not compartmentalized as LWR containments are. Rather, the entire building is open and it is impossible to isolate areas from normal reactor building ventilation flow paths.

Another pertinent feature of the FSV design is that the reactor building is accessible during operation, and maintenance can be, and is, routinely performed during operation. These maintenance activities often involve some limited amount of painting which releases organic solvents to the reactor building atmosphere. Organic compounds are removed from the reactor building atmosphere by charcoal beds in the exhaust filters. These are adsorbed on the charcoal and, if excessive, can reduce its capacity for the adsorption of elemental and methyl iodine. Therefore, it is important to control use of paint in the reactor building and monitor the condition of the charcoal adsorber. In order to assure the operability of the exhaust system, but avoid unnecessary testing, the use of up to 5 gallons of paint can be allowed during a normal 6 month surveillance interval without invoking additional testing. PSC's experience has shown that no significant degradation of the charcoal beds has occurred, even with the use of larger quantities of paint.

This is not considered to involve a significant hazards consideration per 10 CFR 59.92 for the following reasons:

1. The use of up to 5 gallons of paint in the reactor building has been found not to have any significant adverse effect on the iodine removal capacity of the charcoal beds. Therefore, this does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. This involves no change in the design of the plant or the normal or emergency operation of plant systems. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. This does not significantly reduce a margin of safety. Experience has shown that the use of this quantity of paint does not significantly reduce the effectiveness of the charcoal adsorber.

NRC Comment: LCO 3.6.5.3-1 (G-86285)

The prerequisites for testing the reactor building louvers contained in current Technical Specification SR 5.5.2 have been removed from the technical specifications. The prerequisites were provided to limit the conditions when the louvers could be opened for testing as reactor building confinement integrity is compromised. TSUP LCO 3.6.5.1 provides Action time limits for loss of confinement integrity and this accomplishes the same objectives. The prerequisites are procedural in nature and therefore will be included in the implementing procedures for TSUP SR 4.6.5.3.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. The prerequisites are procedural in nature and will be included in the implementing procedure. Furthermore, the SR will be performed within the time limits of LCO 3.6.5.1 and therefore will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. This deletion does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no plant modifications or changes in the way the testing associated with this change will be performed.

3. This deletion does not involve a significant reduction in a margin of safety. The testing will be done within the restoration of operability time limits of the technical specifications and the implementing procedure will ensure the prerequisites are complied with.

NRC Comment: LCO 3.7.1.2-2 (G-86285)

STARTUP has been omitted from the APPLICABILITY of TSUP Technical Specification 3/4.7.1.2 which applies to operability of the steam/water dump system. This is a change from the requirements of current Technical Specification LCO 4.3.3 which apply above 2% power.

The steam/water dump system minimizes water leakage into the core resulting from a steam generator tube rupture. Proper operation of the system minimizes graphite oxidation resulting from the steam-graphite reaction, although safety analyses show no significant deterioration of the fuel or graphite and no excessive PCRV pressure would result from failure of the system to function. The steam-graphite reaction is strongly temperature dependent and insignificant below 900°F. In addition it is highly endothermic and, thus, self-limiting.

Steam generator tube rupture accidents are described in FSAR Section 14.5. The full range of hypothetical scenarios from normal operation to incorrect operation, and, finally, to complete failure of all protective systems have been evaluated for single and multiple tube ruptures at full power. In no case does postulated graphite oxidation or fuel corrosion threaten the ability of core components to perform their design functions.

The average graphite temperature in STARTUP (i.e., up to 5% power) is less than about 500°F. At these temperatures, steam-graphite reaction is insignificant and, therefore, the steam/water dump system is not required to be OPERABLE.

At higher power levels (and average graphite temperatures), the steam/water dump system is required for operation. In the event of steam generator tube rupture, the steam/water dump system would normally be actuated and steam in-leakage would be controlled within 30 seconds.

In the event of a SCRAM from LOW POWER or POWER, the steam/water dump system would remain operable, as it is required in those operational modes, for at least the 30 second time period during which it pe .orms its intended function.

During normal, controlled power reduction from POWER, to LOW POWER, to STARTUP, average graphite temperature would be the same as during power ascension.

Accidents in the STARTUP mode are characterized by small, slow temperature excursions because of the low power level and high heat capacity of the core, and the relatively low temperature of the core components other than fuel. (e.g., CSBs, reflector, etc.). Therefore, graphite oxidation or fuel corrosion are not considered significant safety concerns in the STARTUP mode.

This is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. Since graphite oxidation and fuel corrosion are insignificant in the range of possible temperature and moisture conditions that could exist in the STARTUP mode, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. This change does not create the possibility of a new or different kind of accident from any previously evaluated, because any conceivable accident would be bounded by the analysis of a steam generator tube rupture at 100% power with failure of the steam/water dump system (FSAR Section 14.5.3.3) which concluded that no significant deterioration of the fuel nor excessive PCRV pressure would result from such an accident.

3. This change does not involve a significant reduction in a margin of safety. The steam/water dump system only serves to reduce significantly the primary coolant impurities and the amount of core corrosion following a large steam generator leak. Inasmuch as such an accident has no significant consequences relative to the safety of the plant, its staff, or the public, and operation of the steam/water dump system is not assumed for any limiting accident analysis, no margin of safety concern exists.

NRC Comment: LCO 3.7.1.2-5 (G-86285)

This comment addressed the surveillance requirements for steam water dump system instrumentation. The TSUP deleted the calibration requirements for this instrumentation, as identified in current Technical Specification SR 5.3.1.

This change is justified by the guidance provided in the standard technical specifications (STS). Consistent with the STS, instrumentation that is considered non-critical (i.e., not required directly or indirectly for monitoring core performance or initiating automatic protective actions) is addressed via the plant's administrative controls. Operability is assured by a calibration program that is managed by PSC, and is not explicitly directed by the technical specifications.

The steam water dump system instrumentation is considered non-critical instrumentation and will be included in the FSV instrument calibration program. The reliability of the tank level instrumentation will be ensured as this instrumentation is normally used to verify tank level per SR 4.7.1.2.a.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. The deletion of existing surveillance requirements for steam water dump system instrumentation, and the delegation of these surveillances to plant administrative controls does not involve a significant increase in the probability or consequences of an accident previously evaluated. Instrument operability will be assured in a manner consistent with other nuclear plants that operate per STS requirements.

2. This deletion does not create the possibility of a new or different kind of accident from any accident previously evaluated. The need for operable instrumentation to monitor steam water dump system performance has not been eliminated. This change places the actual calibrations in the plant's administrative controls versus the technical specifications.

3. This deletion does not involve a significant reduction in a margin of safety. The margin of safety for the steam water dump system is in the system design, which is not affected by this change.

NRC Comment: LCO 3.7.1.3-2 (G-86285)

This comment addressed the content of the Design Features discussion on plant safety valves. To eliminate confusion regarding which valves are discussed in the Design Features and which valves are addressed in an LCO, the Design Features discussion was deleted. This involves a deletion of current Technical Specification DF 6.2.3.

The current Technical Specification DF 6.3.2 describes four sets of steam plant safety valves. Only two sets are safety related because they are relied upon in accident analyses. Operability requirements for the safety related main steam and reheater safety valves have been incorporated in proposed Technical Specifications 3/4.7.1.5 and 3/4.7.1.6. The other two sets of steam safety valves protect the bypass flash tank and the hot reheat lines and are for equipment protection only. They are not relied upon in any safety analyses and, therefore, need not be the subject of any technical specification.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The description of safety related design features has been replaced with corresponding operability and surveillance requirements. The deletion of the non-safety related features from the technical specifications has no significant effect on plant design or operation. The non-safety related features were not taken credit for in any accident evaluation.

2. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no plant modifications associated with this change. Safety related features are incorporated in operability and surveillance requirements and non-safety related features were not relied upon in any accident evaluation.

3. This change does not involve a significant reduction in a margin of safety. Incorporation of safety-related design features in operability and surveillance requirements ensures that the required safety margins are maintained.

NRC Comment: LCO 3.7.2-1 (G-86285)

This comment deals with a change in the ACTION statement associated with the loss of hydraulic system pressure to one group of secondary coolant valves. Existing Technical Specification LCO 4.3.7 requires shutdown within one hour and isolation of the non-affected secondary coolant loop. The proposed technical specification requires the affected secondary coolant loop to be isolated within one hour and the plant to be SHUTDOWN within 24 hours.

This change is proposed because design changes make it preferable to cooldown using the non-affected secondary coolant loop, thereby avoiding the plant transient caused by the manual scram necessitated by a one hour ACTION time.

Formerly, the loss of operability of hydraulic assist valves in the emergency feedwater/condensate system required that the non-affected loop be isolated. However, those valves now have motor operators, or manual isolation valves have been added, which allow the operator to isolate the affected secondary coolant loop valves within 1 hour and shut down the reactor in a controlled manner while attempting to recover affected loop hydraulic pressure. Therefore, the urgency to shut down the reactor does not exist as before the design change and it is not necessary to subject the plant to the transients associated with immediate SHUTDOWN.

This feature will be incorporated in an FSAR revision 6 to 18 months after technical specification approval.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. The design change to add motor operators and manual isolation valves that permit safe, controlled shutdown and cooldown assures that this change does not involve a significant increase in the probability or consequences of accidents previously evaluated.

2. This change does not create the possibility for a new or different kind of accident than previously evaluated. The design change makes it possible to isolate a secondary loop affected by loss of hydraulic pressure and to cooldown on the unaffected loop.

3. This change does not involve a significant reduction in margins of safety. It helps to assure controlled core cooling in case of hydraulic system problems and avoids the unnecessary plant transients associated with immediate shutdown.

NRC Comment: LCO 3.7.2-3 (G-86285)

This comment addressed the surveillance requirements for hydraulic system pressure indicators and alarms. The TSUP deleted the calibration requirements for this instrumentation, as identified in current Technical Specification SR 5.3.5.

This change is justified by the guidance provided in the standard technical specifications (STS). Consistent with the STS, instrumentation that is considered non-critical (i.e., not required directly or indirectly for monitoring core performance or initiating automatic protective actions) is addressed via the plant's administrative controls. Operability is assured by a calibration program that is managed by PSC, and is not explicitly directed by the technical specifications.

The hydraulic system pressure indicators and alarms are considered non-critical instrumentation and will be included in the FSV instrument calibration program. PSC will ensure the reliability of this instrumentation, as it is normally used to verify hydraulic pressure greater than 2500 psig, as required by SR 4.7.2.

The deletion of explicit surveillance requirements for the calibration of these instruments is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. The deletion of existing surveillance requirements for hydraulic system instrumentation, and the delegation of these surveillances to plant administrative controls does not involve a significant increase in the probability or consequences of an accident previously evaluated. Instrument operability will be assured in a manner consistent with other nuclear plants that operate per STS requirements.

2. This deletion does not create the possibility of a new or different kind of accident from any accident previously evaluated. The need for operable instrumentation to monitor hydraulic system performance has not been eliminated. This change places the actual calibrations in the plant's administrative controls versus the technical specifications.

3. This deletion does not involve a significant reduction in a margin of safety. The margin of safety for the hydraulic system is in the system design, which is not affected by this change.

NRC Comment: LCO 3.7.3-1 (G-86285)

This comment addressed the surveillance requirements for instrument air system instrumentation. The TSUP deleted the calibration requirements for this instrumentation, as identified in current Technical Specification SR 5.3.6.

This change is justified by the guidance provided in the standard technical specifications (STS). Consistent with the STS, instrumentation that is considered non-critical (i.e., not required directly or indirectly for monitoring core performance or initiating automatic protective actions) is addressed via the plant's administrative controls. Operability is assured by a calibration program that is managed by PSC, and is not explicitly directed by the technical specifications.

The pressure indicators and alarms on the instrument air system receiver tanks and headers are considered non-critical instrumentation and will be included in the FSV instrument calibration program. PSC will ensure the reliability of the receiver instrumentation as it is normally used to verify that the air receiver pressure is greater than 85 psig, per SR 4.7.3.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. The deletion of existing surveillance requirements for instrument air system instrumentation, and the delegation of these surveillances to plant administrative controls does not involve a significant increase in the probability or consequences of an accident previously evaluated. Instrument operability will be assured in a manner consistent with other nuclear plants that operate per STS requirements.

2. This deletion does not create the possibility of a new or different kind of accident from any accident previously evaluated. The need for operable instrumentation to monitor instrument air system performance has not been eliminated. This change places the actual calibrations in the plant's administrative controls versus the technical specifications.

3. This deletion does not involve a significant reduction in a margin of safety. The margin of safety for the instrument air system is in the system design, which is not affected by this change.

NRC Comment: LCO 3.7.6.2-1 (G-86285)

The requirement to declare diesel generators inoperable if CO2 systems are inoperable for 30 or more days was deleted from the existing requirements of Technical Specification LCO 4.10.6 to be consistent with standard technical specification requirements.

Due to the fact that emergency AC electrical power sources are no more critical to prevent or mitigate accidents at FSV than they are at light water reactors, a requirement for fire suppression systems (CO2 systems) for those electrical power sources should be no more restrictive than for comparable light water reactors. Thus, adoption of the standard technical specification requirements for CO2 systems, which is relief from existing requirements, is considered acceptable from a risk perspective.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. The deletion of the requirement to declare the diesel generators inoperable if CO2 systems are inoperable for 30 days does not involve a significant increase in the probability or consequences of an accident previously evaluated. The operability of emergency AC electrical power will be assured in a manner consistent with other nuclear plants that operate per STS requirements. Also, actions to be taken to compensate for the inoperability of CO2 systems are consistent with those required for other STS plants.

2. This deletion does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no plant modifications associated with this change and the compensatory measures required by the specification ensure protection at a level consistent with other STS plants.

3. This deletion does not involve a significant reduction in a margin of safety. The system design and other specified actions provide protection consistent with STS requirements.

NRC Comment: LCO 3.7.6.3-1 (G-86285)

The requirement to shutdown the reactor if halon system operability is not restored in 72 hours was deleted from the existing requirements of LCO 4.10.2, to be consistent with standard technical specification (STS) requirements.

Due to the fact that halon systems are no more critical to prevent or mitigate accidents at FSV than they are at light water reactors, the requirements for fire suppression systems (halon systems) for the control room, auxiliary electric equipment room, and 480 volt switchgear room should be no more restrictive than they are for comparable light water reactors. Thus, adoption of the (STS) requirements for halon systems, which is relief from existing requirements, is considered acceptable from a risk perspective.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. The deletion of the requirement to shutdown the reactor if halon systems are inoperable for 72 hours does not involve a significant increase in the probability or consequences of an accident previously evaluated. The operability of the halon system for the control room, auxiliary electric equipment room, and 480 volt switchgear room will be assured in a manner consistent with other nuclear plants that operate per STS requirements. Also, actions to be taken to compensate for the inoperability of halon systems are consistent with those required for other STS plants.

2. This deletion does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no plant modifications associated with this change and the compensatory measures required by the specification ensure protection at a level consistent with other STS plants.

3. This deletion does not involve a significant reduction in a margin of safety. The system design and other specified actions provide protection consistent with STS requirements.

NRC Comment: LCO 3.7.6.3-2 (G-86285)

The requirement for annual verification of halon system operability by a flow check and test of dampers was changed to an 18 month interval consistent with standard technical specification (STS) intervals. The 18 month interval was chosen for STS allowing for extended fuel cycles and the desire to perform these surveillances while shutdown. Thus, the proposed requirements are acceptable as they are consistent with STS.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. The extension of the test interval from 12 to 18 months does not involve a significant increase in the probability or consequences of an accident previously evaluated. The demonstration of halon system operability at a frequency accepted by the NRC assures system operability at a level consistent with other nuclear plants that operate per the standard technical specifications.

2. This extension does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no plant modifications associated with this change and the halon system surveillances assure operability at a level consistent with other STS plants.

3. This extension does not involve a significant reduction in a margin of safety. The system design and other surveillances assure operability consistent with STS requirements.

NRC Comment: LCO 3.7.6.4-1 (G-86285)

TSUP LCO 3.7.6.4 does not require the installation of a "gated wye" in the event of an inoperable fire hose station, as would be required by the standard technical specifications (STS). Further more, in the event of an inoperable fire hose station, TSUP LCO 3.7.6.4 allows PSC to either ensure that the nearest OPERABLE hose station can provide coverage to the affected area or route additional hose from the nearest OPERABLE hose station. The current LCO 4.10.7 requires that an alternate hose be laid out to the unprotected area from an OPERABLE fire hose station.

The use of gated wyes is not a part of the FSV design or licensing basis and would require a plant change that is beyond the scope of the TSUP. The Action to ensure that the nearest OPERABLE hose station can provide coverage is justified as follows:

The FSV Turbine and Reactor buildings are "open" buildings, having few physical barriers in horizontal or vertical planes. There are four areas that are separated by walls and floors are predominantly open grating. In general, this provides ready access to any given area from several directions for fire fighting purposes.

Manual hose stations are provided throughout the turbine and reactor buildings to provide coverage to all areas. Each hose station is equipped with at least 100 feet of hose and hose stations are separated by no more than 115 feet. The majority of the areas within the turbine and reactor buildings are within reach of two or more hose stations.

The spacing of hose stations, the overlapping coverage, and the open building design, all considered, ensure that any fire area will be protected. If a fire hose station becomes inoperable, it is appropriate to verify coverage from the nearest fire hose station.

This is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. The TSUP specification for fire hose stations is consistent with the Fire Protection Program Plan and the STS guidance. Continued reactor and turbine building fire protection is assured. Therefore, it does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. This does not create the possibility of a new or different kind of accident from any accident previously evaluated because it does not involve any changes in the normal or emergency operation of the plant or in the design function of any plant system equipment or structure. PSC considers that all plausible fire hazards have been evaluated by the plant Fire Hazards Analysis (FHA) and the Fire Protection Program Plan (FPPP). Continued operation per existing design and FPPP requirements does not create any new fire hazard.

3. This does not involve a significant reduction in safety margin because it represents no change from existing design and FPPP requirements.

NRC Comment: LCO 3.7.8-1 (G-86285)

TSUP Specification 3/4.8 4 includes ACM diesel start test intervals that have been revised from the weekly interval required by existing Technical Specification SR 5.20. A test schedule that requires ACM diesel testing either once per 31 days or once per 7 days depending on the number of test failures has been included. PSC has reviewed surveillance data for the 20 week period from November 21, 1987 to April 11, 1988. This review showed 20 successful start and 2 hour load tests, confirming that the reliability of the ACM diesel justifies monthly testing. Furthermore, the TSUP ACM diesel test schedule is similar to that for the standby diesel generators.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the revision in the start test interval provides adequate assurance of the operability of the ACM diesel. The revised schedule permits less frequent tests provided that previous testing demonstrates a high degree of reliability. This reduces the wear and tear associated with testing and, in the long run, contributes to the reliability of the ACM diesel. Should testing show two or more failures per twenty starts, the test interval would be shortened to 7 days until the higher degree of reliability has been restored. ACM diesel operability will be assured in a manner consistent with the standby diesel generators and with the standards at other nuclear plants that operate per STS requirements.

2. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated because it does not involve any changes in the normal or emergency operation of the plant, or in the design function of any plant system, equipment, or component.

3. This change does not involve a significant reduction in a margin of safety. The margin of safety provided by this system is inherent in the design of the system, which is not changed, and in the reliability of the ACM diesel which is confirmed on a basis consistent with other nuclear plant that operate per STS requirements.

NRC Comment: LCO 3.7.9-6 (G-86285)

PSC has established surveillance testing for laboratory measurements of the efficiency of the charcoal adsorber material in the control room emergency ventilation system. The testing and acceptance criteria for the system generally follow the form of the standard technical specifications (STS).

This comment required clarification of the basis for determining methyl iodide penetration requirements of this technical specification.

The methyl iodide penetration criteria has been changed to 3 which is the criteria given in ANSI N509-1980, Table 5-1. This change is consistent with NRC comments on Control Room Habitability Requirements in NRC letter, Heitner to Williams, dated 11/24/86 (G-86613) and PSC's response in PSC letter, Brey to Calvo, dated May 15, 1987 (P-87133). This methyl iodide penetration criteria is also consistent with the equipment design specification, 75-F-03.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the surveillance testing helps assure that the control room emergency ventilation system will perform at or above efficiencies specified in the equipment design specifications.

2. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated because it does not involve any changes in the normal or emergency operation of the plant, or in the design function of any plant system, equipment, or component.

3. This change does not involve a significant reduction in the margin of safety. The margin of safety provided by this system is inherent in the design of the system, which is not changed, and in the performance of the control room emergency ventilation system which is confirmed on a basis consistent with industry and regulatory guidelines.

NRC Comment: LCO 3.7.10-1 (G-86285)

This comment addressed the applicability of the snubber technical specification requirements. The requirement of current LCO 4.3.10 for snubbers above 2% power was changed to 5% power in the TSUP, consistent with the TSUP definition of LOW POWER.

Analyses documented in FSAR Appendix D.4.2 show that in the unlikely event of a Loss of Forced Cooling from an 8% equilibrium reactor power level, fuel temperatures do not approach 2900°F even with no systems operating that would utilize snubbers. Also, per FSAR Appendix D.4.1, cooldown from a 35% equilibrium power level is acceptable using only the PCRV Liner Cooling System, which uses no snubbers.

Furthermore, the basic energy parameters such as core temperatures, pressures, and decay heat levels are not significantly different between 2% power and 5% power. This allowance to not include STARTUP in the Applicability of this Specification is required to permit core dry-out and startups for training purposes.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. There are no significant differences between 2% and 5% as to the total potential energy release (as shown by basic energy parameters) therefore this change does not involve a significant increase in the probability or consequences of an accident previously evaluated. Snubber operability in the systems required for the particular mode of operation will be assured in a manner consistent with the FSAR requirements. Also, actions to be taken to compensate for the inoperability of snubbers are consistent with those required for other STS plants.

2. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no plant modifications associated with this change and the compensatory measures required by the specification ensure protection at a level consistent with other STS plants.

3. This change does not involve a significant reduction in a margin of safety as there are no significant differences in potential energy release that could impact plant safety. The specified actions provide protection consistent with STS requirements.

NRC Comment: LCO 3.7.10-6 (G-86285)

TSUP SR 4.7.10 requires a functional test of 10% of each type of snubber and, for each snubber that does not meet the acceptance criteria, an additional 5% must be tested. Current Technical Specification SR 5.3.8 requires an additional 10%.

This change is justified as it is consistent with the latest draft of ASME OM-4 Industry Standard, "Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)". This position has also been accepted for other nuclear plants such as Fermi, with no site-specific issues that would not apply to Fort St. Vrain.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. Reducing the size of the snubber population to be tested after a snubber does not meet the acceptance criteria does not significantly increase the probability or consequences of an accident previously evaluated. The reduction from 10% to 5% is consistent with the ASME OM-4 Industry Standard, as has been implemented at other nuclear power plants.

2. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. There is no change in the plant design, equipment, or in the normal or emergency operation of plant systems.

3. This change does not involve a significant reduction in a margin of safety, as the operability requirements for plant snubbers are not changed. Also, the degree of reliability is consistent with recognized industry guidance.

NRC Comment: LCO 3.8.1-5.7 (Attachment 1, P-87161)

This comment is repeated as comments 2 and 15 in Attachment 2 to P-87161.

The current Technical Specification LCO 4.6.1.g requires that the auxiliary boiler(s) be shutdown upon reaching a minimum diesel fuel oil level of 20,000 gallons. This requirement is not retained in TSUP LCO 3.8.1, and is justified as follows:

The action to shutdown auxiliary boilers in the event of low fuel oil level is intended to conserve the remaining oil for use in the standby diesel generators (SDG). A large part of the fuel oil storage is shared between this equipment. The TSUP LCO accomplishes the same objective by identifying that 20,000 gallons of diesel fuel in underground storage (including 5500 gallons in the dedicated diesel fuel oil storage tank) are required for SDG operability. This assures enough fuel for one SDG for at least one week, under required load conditions to shutdown the plant and maintain it in a safe condition.

The actions regarding the auxiliary boiler are overly prescriptive and inconsistent with the guidance provided in the Standard Technical Specifications (STS). Safe plant operation is achieved by compliance with the TSUP LCO 3.8.1 requirements or by following the applicable Action requirement.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. The deletion of a requirement to shutdown the auxiliary boiler(s) when stored diesel fuel oil level is less than 20,000 gallons does not significantly increase the probability or consequences of an accident previously evaluated. SDG operability depends on at least 20,000 gallons of diesel fuel in underground storage and appropriate actions are required with less than this requirement. SDG operability is assured in a manner consistent with other plants with STS requirements.

2. This change does also not create the possibility of a new or different kind of accident from any accident previously evaluated. There is no change in the plant design, equipment, or in the normal or emergency operation of plant systems.

3. This change does not involve a significant reduction in a margin of safety. The ability to supply required loads for at least one week with a single SDG is retained.

NRC Comment: LCO 3.9.3-2 (G-86285)

This comment originally addressed the fact that TSUP SR 4.9.3.c extended the channel calibration requirement of current SR 5.7.2 from 12 to 18 months. This includes the instrumentation for fuel storage facility helium pressure and cooling water flow and temperature.

Subsequent to the identification of the comment, PSC has deleted the explicit requirements for surveillance of this instrumentation, consistent with the TSUP position on non-critical instrumentation. Also, the applicable specification has been re-designated as 3/4.9.4. This justification addresses the deletion of the surveillances of this instrumentation, as opposed to the extension of the surveillance interval.

The deletion of the surveillance is justified by the guidance provided in the standard technical specifications (STS). Consistent with the STS, instrumentation that is considered non-critical (i.e., not required directly or indirectly for monitoring core performance or initiating automatic protective actions) is addressed via the plant's administrative controls. Operability is assured by a calibration program that is managed by PSC, and is not explicitly directed by the technical specifications.

The fuel storage facility helium pressure indicators and alarms, the cooling system flow indicators and alarms, and the cooling system temperature alarms are considered non-critical instrumentation. PSC will ensure the reliability of this instrumentation via the FSV calibration pr gram, as it is normally used to verify the flows and temperatures identified in SR 4.9.4.

The deletion of explicit surveillance requirements for the calibration of these instruments is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. The deletion of existing surveillance requirements for fuel storage facility instrumentation, and the delegation of these surveillances to plant administrative controls does not involve a significant increase in the probability or consequences of an accident previously evaluated. Instrument operability will be assured in a manner consistent with other nuclear plants that operate per STS requirements.

2. This deletion does not create the possibility of a new or different kind of accident from any accident previously evaluated. The need for operable instrumentation to monitor fuel storage facility performance has not been eliminated. This change places the actual calibrations in the plant's administrative controls versus the technical specifications.

3. This deletion does not involve a significant reduction in a margin of safety. The margin of safety for the fuel storage facility is in the system design, which is not affected by this change.

NRC Comment: LCO 3.9.3-3 (G-86285)

TSUP LCO 3.9.4.b requires that the fuel storage well emergency booster fan must be capable of moving a minimum total air flow of 9,000 cfm through the fuel storage facility. The existing Technical Specification LCO 4.7.3.b requires 12,000 cfm; 9,000 cfm for the effected vault and 1,500 cfm for each of the two other vaults. This change is in accordance with FSAR analyses which state 9,000 cfm would provide adequate cooling assuming failure of both water cooling loops. FSAR Section 9.1.2.3 (Rev. 5) no longer implies 12,000 cfm is required and clearly states 9,000 cfm will prevent excessive temperatures.

The analysis in FSAR Section 14.6.3.2 (Rev. 5) and in General Atomic Report GAMD-7346 is based on a conservative assumption that the ventilation rate would be 9,000 cfm distributed equally between the vault compartments. The vent system, through the use of dampers, could be lined up to connect only one vault compartment (the one with the loss/interruption of cooling) thereby increasing the cooling rate. The calculations show that the maximum fuel temperature would reach a peak of 2200°F, well below the temperature at which significant damage to the fuel particles occurs, and no other temperatures would result in a loss of any safety function. The analysis shows that adequate time is available to take corrective actions but if no actions were taken there would still be no uncontrolled release of activity to the atmosphere since the fission products being released from the fuel would be vented to and collected by the gas waste system.

This change is not considered to involve a significant hazards consideration per 10CFR50.92, for the following reasons:

1. Even if shared equally, 9,000 total cfm through the three vault compartments supplied by the emergency booster fan is sufficient to ensure temperatures do not result in significant fuel damage or loss of safety functions. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no plant modifications associated with this change and current analyses encompass this change.

3. This change does not involve a significant reduction in a margin of safety. The system design has not been changed and the FSAR analyses support this change in required air flow.

NRC Comment: LCO 3.9.3-8 (G-86285)

An ACTION has been added to TSUP Technical Specification LCO 3.9.4 to permit performance of an engineering evaluation as an alternative to other required actions in case of the loss of fuel storage well cooling capacity. This is justified by the fact that cooling capacity requirements for the fuel storage facility are based on storage of fuel with the maximum decay heat possible, that is, fuel just removed from the reactor after steady full power operation for several months. The actual decay heat source may be significantly less depending on the power level prior to shutdown and the decay time since shutdown. For example, the calculated adiabatic heatup rate of a fuel element 100 days after removal from the reactor that had been at full power, is only about 1 degree F per hour. Therefore, an engineering evaluation can be conservatively used to determine at what time, if ever, a fuel element surface temperature of 750 degrees F would be reached and, thus, determine the appropriate action.

This change is not considered to involve a significant hazards consideration per 10CFR50.92 for the following reasons:

1. Permitting an engineering evaluation as an alternative to other required actions in case of loss of fuel storage well cooling capacity does not involve a significant increase in the probability or consequences of an accident previously evaluated. An evaluation that considers the actual decay heat rate of fuel in the affected storage well, the actual heat removal capacity of the cooling system, if any, and the thermal capacity of the fuel could conservatively determine how long the stored fuel surface temperature would remain below 750 degrees F, a temperature at which graphite oxidation is not a significant concern.

2. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no plant modifications associated with this change.

3. This change does not involve a significant reduction in a margin of safety. Performing an engineering evaluation to conservatively determine projected stored fuel temperatures provides an acceptable basis for determining the nature and timing of corrective actions that prevent graphite oxidation.

NRC Comment: AC 6.5.1.6.e (G-86285)

TSUP AC 6.5.1.6.e changes PORC's responsibility from "performance" to "review" of investigations of all violations of the technical specifications, including the preparation and forwarding of reports covering the evaluation and recommendations to prevent recurrence to the Manager, Nuclear Production, and to the Chairman of the NFSC. This is a change from current Technical Specification AC 7.1.2.5.e. PORC will retain the responsibility to ensure that an investigation occurs when there is a technical specification violation, however, the actual performance of the investigation will normally be accomplished by a FSV staff organization, and reviewed by PORC to provide independent verification. This is consistent with current practice and it assures thorough review.

This change is not considered to involve a significant hazards consideration per 10CFR50.92, for the following reasons:

1. This change in wording clarifies who performs the investigation and doesn't change PORC's responsibility to ensure an investigation is performed. Further, there is no change to the material that will be covered by the investigation and report. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. This change in wording does not create the possibility of a new or different kind of accident from any accident previously evaluated because there are no plant/design changes associated with this change.

3. This change in wording does not involve a significant reduction in a margin of safety. Violations of the technical specifications are still investigated and the report contains the same information as before.

NRC Comments: AC 6.9-2 and AC 6.9-4 (G-86285)

TSUP Technical Specification 6.9 has been rewritten consistent with the W-STS Rev. 5. It also includes all the existing requirements in AC 7.5. PSC has reviewed the Radiological Environmental Technical Specifications (RETS) and the existing AC 7.5 to confirm that all requirements of AC 7.5 have been incorporated into the proposed AC 6.9 and that there is no duplication between the RETS and the proposed AC 6.9.

PSC considers that the reports required by existing AC 7.5.3 have been incorporated into AC 6.9.1 through 6.9.15 and no requirements have been deleted. Therefore, no analysis for a significant hazards consideration per 10 CFR 50.92 is required.

Figure 3.2.2-1 (No NRC Comment)

TSUP Figure 3.2.2-1 identifies the allowable mismatch between the individual region outlet temperatures and the core average outlet temperature. This figure is included in current Technical Specification Figure 4.1.7-1.

Current Figure 4.1.7-1 does not include the full range of operating parameters experienced at FSV, and it has historically been extrapolated to determine applicable limits. While this extrapolation is justifiable, TSUP Figure 3.2.2-1 has been expanded to encompass all allowable operating ranges, and to identify those conditions where operation is restricted.

The Average Core Temperature Rise range has been expanded from 660 - 755°F to 400 - 800°F. The Circulator Inlet Temperature range has been expanded from 490 - 750°F to 450 - 750°F. The mismatch limit curves were extrapolated to the extent required, while maintaining the restriction that region outlet temperature cannot exceed 1555°F, consistent with FSAR Table 3.6-1 and the basis for current LCO 4.1.7. This extrapolation retains the same methodology and conservatisms included in the development of Figure 4.1.7-1, per GA Topical Report GA-C16781.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. The extrapolation of Figure 4.1.7-1 to include the full range of operating parameters encountered during FSV operation does not significantly increase the probability or consequences of an accident previously evaluated. The methodology and conservatisms assumed during the development of Figure 4.1.7-1 are retained. Operation per TSUP Figure 3.2.2-1 will keep maximum fuel temperatures within FSAR stated values, regardless of the power level or the amount of core bypass flow which may exist.

2. This change also does not create the possibility of a new or different kind of accident from any accident previously evaluated. There is no change in plant design, equipment, or in the normal or emergency operation of plant systems.

3. This change does not involve a significant reduction in a margin of safety. The margins included in the current mismatch limits have been retained.

AC 6.5.2.10.k and AC 6.8.1.i (not identified in written correspondence)

TSUP Specification 6.5.2.10.k requires an NFSC audit of the QA Program for effluent and environmental monitoring, and TSUP Technical Specification 6.8.1.i requires procedures for the QA Program for effluent and environmental monitoring. Neither of these specifications refer to Regulatory Guides 1.21 and 4.1 for the QA Program, as is the case for current Technical Specifications AC 7.1.3.8.c.11 and AC 7.4.a.8.

The reference to Regulatory Guides 1.21 and 4.1 is included in the Quality Assurance Program for Plant Operation, FSAR Appendix B, so their inclusion in the technical specifications in unnecessary. The deletion of these references is also consistent with the corresponding requirements of the STS. This is considered an editorial change that has no effect on plant operation or safety.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated. It is an editorial change only and has no effect on plant operation or safety.

2. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated because it is editorial only.

3. This change does not involve a significant reduction in a margin of safety. The relevant regulatory requirements are incorporated in the Quality Assurance Program for Plant Operation, so this change has no effect on plant operation or safety.

Deletion of Surveillance Requirements for the Breathing Air System

Surveillance requirements contained in the current SR 5.10.5 which require functional testing and air quality testing of the breathing air system have not been included in the TSUP.

The breathing air system is designed to provide a continuous supply of purified air to air-hose line type respirators in the Control Room, Auxiliary Electric Equipment Room, and the 480V Switchgear Room. The system is intended to allow the Fire Brigade personnel to enter these areas for firefighting activities. The system can also be used by operating personnel for reactor operation activities during conditions when the room air could be potentially dangerous to health.

In accordance with 10 CFR 50.36, ANSI/ANS-58.4-1979, and the work specification for the Technical Specification Upgrade Program (WS-TS-1) the technical specifications are derived from the evaluations and analyses included in the FSAR. Furthermore, the technical specifications contain only those items specifically required by Federal Regulations.

Credit is taken for the breathing air system as a supplement to the self-contained breathing apparatus system, but neither of the breathing air systems will be necessary should the plant's fixed suppression systems and associated HVAC systems be OPERABLE. The control room emergency ventilation system is relied upon to ensure control room habitability.

The FSAR, 10 CFR 50.36, App. R, Reg Guide 1.120, and other associated documents were reviewed to determine if an LCO and an associated Surveillance Requirement (SR) are necessary for the plant's breathing air system. None of these documents nor the Standard Technical Specifications require a condition or limitation upon reactor operation with respect to the breathing air system.

10 CFR 50 App. R specifies that surveillance procedures be established to ensure that fire barriers are in place and that fire suppression systems are operable. It does not specify that the breathing air system and other Fire Brigade protective equipment (i.e., hard hats, emergency communications equipment, portable extinguishers, etc.) be included in the technical specifications.

In summary, the breathing air system is not relied upon in the Fort St. Vrain safety analyses nor is it required by Federal regulations to be included in the technical specifications. Furthermore, the breathing air system does not fall under the "immediate threat" standard (1). Consequently, the surveillance requirement on the breathing air system has been deleted from the FSV Technical Specifications and will instead be included within the plant's Administrative Controls Program.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. The deletion of surveillance requirements for the breathing air system does not involve a significant increase in the probability or consequences of an accident previously evaluated. The system is not relied upon in any safety analyses or evaluation.

2. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The relevant surveillance requirements will be included in the plant's Administrative Controls Program so this change has no practical effect.

3. This change does not involve a significant reduction in a margin of safety. Since the breathing air system is not relied upon in any safety analysis or evaluation, it is not considered to provide any significant margin of safety. Furthermore, this change has no practical effect because the plant design and operation are not affected and the surveillances will be continued under the Administrative Controls Program.

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(1). The Atomic Safety and Licensing Appeal Board has propagated an "immediate threat" standard for defining what should be included in the technical specifications. IN ALAB-531, the Board stated that: "... as best we can discern it, the contemplation of both the act and the regulations is that Technical Specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." (In the Matter of Portland General Electric Company, et al. (Trojan Nuclear Power Plant), 9 NRC 263 (1979).)

Deletion of Core Delta P Technical Specification

The core delta P technical specification (formerly SR 5.4.6 - Core Delta P Indicator) has been deleted from the Technical Specification Upgrade Program.

The core delta pressure indicator (PDT-1112) was used to monitor core pressure drop during rise-to-power tests and fluctuation testing. This pressure indicator is not part of the plant protective system, the plant's regulating system, nor is it part of the plant's nuclear instrumentation.

In accordance with 10 CFR 50.36, ANSI/ANS-58.4-1979, and the Work Specification for the Technical Specification Upgrade Program (WS-TS-1) the technical specifications are derived from the evaluations and analyses included in the FSAR. Furthermore, the technical specifications shall contain only those items relied upon in the safety analyses and/or those items specifically required by Federal Regulations.

The core delta pressure indicator is not relied upon in the Fort St. Vrain safety analyses/evaluations. Credit is taken for the instrument in FSAR Subsection 7.3.3.2 as an indicator of the total core pressure drop. However, the indicator is not used in response to any of the accident analyses. During normal operation, fluctuations in core differential pressure are reflected by the region outlet temperature thermocouples (Specification 3/4.2.2).

The FSAR, 10 CFR 50.36, 10 CFR 50 Appendix A, Reg Guide 1.97 and other associated documents were reviewed to determine if an LCO and an associated Surveillance Requirement (SR) are necessary for the core delta pressure indicator. Neither of these documents nor the Standard Technical Specifications require a condition or limitation upon reactor operation with respect to an inoperable core differential pressure indicator.

In summary, the core Delta P indicator is not relied upon in the Fort St. Vrain safety analyses nor is it required by Federal Regulations to be included in the technical specifications. Furthermore, the indicator does not fall under the "immediate threat" standard (1). Consequently, the Surveillance Requirement SR 5.4.6 will be deleted from the FSV Technical Specifications and will be appropriately included within the plant's Administrative Controls Program.

This is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. Deletion of the Core Delta P indicator technical specification requirement does not involve a significant increase in the probability or consequences of an accident previously analyzed because it is not relied upon in any safety analyses/evaluations.

2. This change does not create the possibility of a new or different kind of accident from any previously evaluated because this instrument is not part of the plant's regulating system, nuclear instrumentation, or protective systems. It plays no significant role in the prevention or mitigation of accidents.

3. This change does not involve a significant reduction in a margin of safety because the core delta P instrumentation makes no significant contribution to any margin of safety.

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(1). The Atomic Safety and Licensing Appeal Board has propagated an "immediate threat" standard for defining what should be included in the technical specifications. IN ALAB-531, the Board stated that" "____ as best we can discern it, the contemplation of both the act and the regulations is that Technical Specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." (In the Matter of Portland General Electric Company, et al. (Trojan Nuclear Power Plant), 9 NRC 263 (1979).)

Deletion of Fuel Loading and Initial Rise to Power LCO's

The fuel loading and initial rise to power limiting condition for operation (formerly LCO 4.9.1) has been deleted from the Technical Specification Upgrade Program.

LCO 4.9.1 addressed two phases of the initial power ascension test program. Phase 1 included fuel loading and low power physics testing in an air or helium environment. Phase 2 included hot physics tests with helium environment and rise to full power testing.

These low power physics tests began in January, 1974 and were completed in April, 1975. These tests were a prerequisite for the rise-to-power tests which began in April, 1975.

Based on the original technical specification requirement that this test be a "one-time" initial low power physics test and having satisfactorily completed the testing in April 1975, LCO 4.9.1 is eliminated. REFERENCE: PSC letter: J. Gahm to E. H. Johnson dated March 5, 1985 (P-85063).

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92 for the following reasons:

1. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated. Since the LCO only applied to early plant operation and its requirements have been satisfied, it no longer serves a useful purpose and has no affect on the operation or safety of the plant.

2. This change does not create the possibility of a new or different kind of accident from any previously evaluated because the LCO is no longer effective and serves no useful purpose regarding future operation of the plant.

3. This change does not involve a significant reduction in a margin of safety. The LCO has served its intended purpose and is no longer effective or useful.

Revision of Surveillance Intervals to be Consistent with Corresponding STS Intervals

In general, TSUP surveillance intervals have been brought into alignment with corresponding STS surveillance intervals. In some cases, this has involved expanding the current intervals, for example, extending annual calibrations to once per 18 months. The use of STS surveillance intervals, where appropriate for Fort St. Vrain, provides the same degree of assurance and verification of technical specification compliance as is required of other nuclear power pl nts.

This change is not considered to involve a significant hazards consideration per 10CFR50.92 for the following reasons:

1. The extension of surveillance intervals to be consistent with the STS intervals does not involve a significant increase in the probability or consequence of an accident previously evaluated. Equipment operability is assured in a manner consistent with other nuclear plants that operate per STS requirements.

2. This change in surveillance intervals does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no plant modifications associated with this change and no change to plant operations.

3. This change in surveillance intervals does not involve a significant reduction in a margin of safety. The margin of safety for the affected equipment is inherent in the system designs, which are not affected by this change.

Instrumentation Deletions (No associated NRC Comment)

Calibration and testing of non-critical instrumentation that is associated with controls for various plant equipment has been deleted from the technical specifications. Instrumentation associated with core performance and automatic protective action is considered critical instrumentation and the corresponding technical specification surveillances have been retained in the proposed upgraded technical specifications. However consistent with the standard technical specification requirements, non-critical instrumentation surveillances are not explicitly included in the upgraded technical specifications. These non-critical instruments are included in an administratively controlled calibration program.

This change is not considered to involve a significant hazards consideration per 10 CFR 50.92, for the following reasons:

1. The deletion of existing surveillance requirements for non-critical instrumentation, and the delegation of these surveillances to plant administrative controls does not involve a significant increase in the probability or consequences of an accident previously evaluated. Instrument operability will be assured in a manner consistent with other nuclear plants that operate per STS requirements.

2. This deletion does not create the possibility of a new or different kind of accident from any accident previously evaluated. The need for operable non-critical instrumentation has not been eliminated. This change places the actual calibrations in the plant's administrative controls versus the technical specifications.

3. This deletion does not involve a significant reduction in a margin of safety. The margin of safety for non-critical instrumentation is in the respective system design, which is not affected by this change.

Instrumentation for which explicit surveillance requirements have been deleted from the FSV Technical Specifications are listed on the following pages of this Attachment.

INSTRUMENTATION SURVEILLANCES DELETED IN TSUP

Current SR	Instrumentation
5.2.1.c)1)	Pressure switch and alarm for each interspace between rupture disk and the safety valve. Pressure switch and alarm for safety valve tank.
5.2.1.c)2)	Position indication circuits for PCRV overpressure protection shutoff valves. Pressure switch and alarm for PCRV safety valve bellows.
5.2.7.d)	Instruments and controls for turbine water removal tank overflow.
5.2.8.c)	Instruments and controls for Bearing Water Makeup Pumps.
5.2.8.d)	Instruments and controls for Bearing Water Pumps.
5.2.9	Helium Circulator bearing water accumulators instruments and controls.
5.2.10.a)1)	Instruments and controls for motor and engine driven fire pumps.
5.2.15	Instrumentation which monitors delta p between purified Helium supply and PCRV penetration interspaces and primary coolant.
5.2.16.b)	Instrumentation monitoring PCRV penetration interspace gas flows.
5.2.16.c)	Instrumentation monitoring pressure in core support floor and columns.
5.2.16.d)	Controls, position indication, etc. for remote manual isolation valves for pressurizing, purging, venting PCRV closures.
5.2.23	Fire Water Booster Pump instruments and controls.
5.2.24.a)	Pond level instrumentation.
5.2.24.b)	Circulating Water Makeup pump instruments and controls.

Current SR	Instrumentation
5.2.24.e)	Service Water Pump instruments and controls.
5.2.24.f)	Reactor Plant Cooling Water Pump instruments and controls.
5.2.24.g)	Purification Cooling Water Pump instruments and controls.
5.2.24.h)	Instruments used for auto isolation of purification cooling water system and Reactor Plant Cooling Water System.
5.3.1.b)	Steam/water dump tank level indicators.
5.3.1.c)	Steam/water dump tank level, pressure, temperature instruments.
5.3.5	Pressure indicators/low pressure alarms on hydraulic oil accumulators.
5.3.6	Pressure indicators/low pressure alarms on instrument air receiver tanks and headers.
5.3.10a) 5.3.10b) 5.3.10c)	Instrumentation to control emergency condensate flow to reheaters, to automatically open reheater discharge bypass on high pressure, and to monitor reheater discharge bypass temperature and reheater inlet temperature.
5.4.4	PCRV Cooling Water System temperature scanner.
5.4.5	PCRV Cooling Water System flow scanner.
5.4.6	Core Delta P instrumentation.
5.4.7	Control Room temperature control thermostat.
5.4.11	PCRV Surface Temperature indicator.
5.4.13	480V Switchgear room temperature indicator and alarm.
5.5.1	Instrumentation monitoring Reactor Building pressure.
5.5.2	Reactor Building overpressure Delta P switches.
5.7.1.a)	FHM cooling water leak detector.

Current SR	Instrumentation
5.7.2.a)	Fuel Storage Facility Helium pressure indicators and alarms.
5.7.2.b)	Fuel Storage Facility cooling system flow indicators and flow and temperature alarms.
5.10.6.d)	Reactor Plant Exhaust Filter temperature instruments and controls.