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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENT NO. 28 TO FACILITY OPERATING LICENSE NO. DPR-34 PUBLIC SERVICE COMPANY OF COLORADO FORT ST. VRAIN NUCLEAR GENERATING STATION DOCKET NO. 50-267

1.0 Introduction

Fort St. Vrain, a 330 MWe high temperature gas-cooled reactor (HTGR), was designed by the General Atomic Company (GAC) and is operated by the Public Service Company of Colorado (PSCo) near Platteville, Colorado. PSCo was issued a construction permit on September 17, 1968 and submitted the Final Safety Analysis Report as Amendment 14 to its application for a construction permit and operating license for the Fort St. Vrain Nuclear Generating Station (FSV) on November 4, 1969. A Safety Evaluation Report (SER) dated January 20, 1972 and a first supplement which was issued on June 12, 1973 concluded that FSV can be operated, as proposed, at power levels up to 842 MWt, full 100 percent power, without endangering the health and safety of the public.

After issuing the 1972 SER, several deficiencies were found which, in later years, limited the power level at which Fort St. Vrain could be operated. These were:

1. In addition to the restrictions imposed by the Technical Specifications on Fuel Loading and Initial Rise to Power, a 2 percent hold was imposed on FSV due to a cracked pelton wheel by Commission letter dated November 4, 1974. The "A" helium circulator was removed from its normal position in the PCRV bottom head penetration and replaced with a spare due to problems with the static seal bellows. Removal of the circulator made possible a detailed examination of the circulator components. A fluorescent penetrant examination of the pelton wheel coupling area indicated uniform cracking at the root of all the coupling teeth and revealed six out of twenty pelton turbine buckets with cracks at the root of the splitter. A commitment by PSCo to replace the cast pelton wheel with a forged pelton wheel and to decrease the speed of the pelton wheel from 10,500 rpm to 7,000 rpm resulted in a SER granting continuation of rise-to-power testing above the previously imposed 2 percent limit.

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2. In April 1975, shortly after the occurrence of an electrical cable fire at the Browns Ferry plant, an inspection revealed that some fire stops in the electrical cable system had not been installed and that routing of some cables deviated from the installation criteria set forth in the FSAR. By letter dated June 17, 1975 PSCo stated that FSV would be maintained in a subcritical condition, a condition it had been at since May 1, 1975, until the problem is resolved to the satisfaction of the NRC. The scope of consideration was broadened beyond electric cable segregation and separation to include fire prevention, detection and suppression, and alternate methods for accomplishing orderly plant shutdown and cooldown in case of loss of normal and preferred systems.

   On June 18, 1976 the Commission issued Amendment No. 14 to the license approving all proposed corrective actions and operation of FSV up to a power level of 40 percent of rated power. The Commission determined that "upon completion of all Stage 1 corrective actions, the FSV plant will achieve the safety objective and provide the same margins of safety that were previously found acceptable for full power operation".

3. During Phase 1 operations, testing and operational experience indicated that certain changes in the plant should be made to improve operational reliability and safety; it is unlikely that these items would have been discovered without such experience. In addition, PSCo and General Atomic identified, by letter dated March 1, 1977, an inconsistency between the Facility Technical Specifications and the FSAR, performed accident analyses related to this matter, and indicated that they be limited to 70 percent of rated power to remain within the conditions described in the FSAR until the matter is further resolved. Three items were subsequently identified by letter to PSCo as constituting a 70 percent hold on further reactor operation: (a) accident reanalysis using correct power-to-flow ratios, (b) moisture injection tests and response times, and (c) time available before depressurization is necessary following loss of forced circulation (LOFC). These three items were discovered to be inconsistent with the FSAR and were addressed in meetings and requests for revisions to the Technical Specifications. The staff reviewed the proposed revisions and concluded that they constitute corrections that result in data and analyses consistent with the FSAR. Therefore, all requirements for continued rise-to-power have been satisfied.

4. During the initial rise-to-power program of the FSV reactor in October 1977, while approaching 60% power, temperature fluctuations were observed in the primary coolant circuit at the outlet of individual core regions and the inlet to steam generator modules. A compre-

hensive program of investigation into the nature and cause of the temperature fluctuations was initiated immediately. The fluctuation investigations led to the design and fabrication of region constraint devices (RCDs) as a solution to the problem. These mechanical links were installed at the top of the core in November 1979, at locations where three regions intersect and were designed to provide inter-region linking to stabilize the gaps between regions at the top of the core to near nominal values.

Steady-state testing performed during initial operation following installation of the RCDs verified that the overall core performance was unaffected by the presence of the RCDs. Testing to evaluate the success of RCDs as a solution to the temperature fluctuations was first performed in November and December 1980. These tests confirmed that the RCDs were successful at preventing fluctuations up to 70% power and a core pressure drop of 4.2 psid. However, once in November and again in December, at a transient peak core pressure drop of 3.8 psid, following an increase in reactor power, a region outlet temperature redistribution was observed. These redistributions resulted in the decrease of several boundary region outlet temperatures, particularly in the NW sector of the core, while inner core region outlet temperatures generally increased somewhat more than would be expected from the power increase.

In Amendment No. 23, dated March 16, 1981, the NRC approved testing of FSV above 70 percent power. Testing to confirm the success of the RCDs as a solution to the temperature fluctuations and to investigate the region outlet temperature redistribution above 70 percent power was conducted during March, April, May, October and November of 1981. No fluctuations occurred, but redistributions similar to those experienced in November and December of 1980 were observed.

By application dated July 6, 1982, the licensee requested changes to the TS to: (1) define individual refueling region outlet temperatures, (2) define a comparison region, (3) limit the maximum mismatch between region outlet temperature and core average outlet temperature, and (4) add a new surveillance requirement to assure that the limit on comparison region peaking factor (RPF) discrepancy is met and the values of RPF used to determine the outlet temperature of the seven NW boundary regions are correct.

The reactor achieved criticality on January 31, 1974, and low power physics testing was initiated. These low power tests, identified as the "A Series" tests, along with the "B Series", or power ascension, tests were reported in accordance with Section 7 of the Technical Specifications.

Also, in accordance with the Technical Specification, Public Service Company of Colorado provides "Reportable Event" reports and "Unusual Event" reports of safety items related to abnormal, unusual or unanticipated events that occur during the course of plant operation. The proposed amendment is based on updated data and analyses which have been performed during a test program up to 100 percent power, as authorized in Amendment 23. The test program has confirmed safe operation of the plant up to full power, and the revised Technical Specifications were submitted to assure that safe operation of the plant is maintained. The original fluctuation problem has been resolved and the corrections confirmed by a test program up to full power. In addition to the reports received from the licensee, the NRC staff reviews have benefitted from information on plant status and operations provided by the Office of Inspection and Enforcement, and by visits to the plant site by technical specialists to review plant records and the "as-built" condition of the plant. Our safety review has also included consideration of comparable light water reactor safety under the sponsorship of the Office of Nuclear Regulatory Research and information developed during the review of the General Atomic Standard Safety Analysis Report, GASSAR.

The operating license, DPR-34, was issued on December 21, 1973 and has been amended twenty-eight times including the amendment supported by this safety evaluation. A listing and brief description of the twenty-seven prior amendments is presented in Appendix A.

2.0 Fluctuation Testing

2.1 Summary

After the initial discovery of temperature fluctuations in October 1977, a testing program was established to induce and observe the phenomenon in a controlled manner. The test program, designated as RT-500, was performed in three stages: Cycle 1 was testing prior to installation of the region constraint devices (RCD) and below 70 percent power; Cycle 2 was testing with the RCDs installed and power levels up to 70 percent; and Cycle 3 testing was performed with RCDs and up to 100 percent power. The testing program showed that no fluctuations were present when the RCDs were installed, even in operating regimes where fluctuations occurred prior to installation of the region constraint devices. However, as reactor power was increased rapidly (approx. 3% per minute) in small (approx. 3% power) steps from 40% to 100% power, one or more region outlet temperature redistributions were observed, generally at core pressure drops between 3.7 and 4.0 psid. The temperature redistributions resulted in the decrease of several boundary region outlet temperatures, particularly in the NW sector of the core, while inner core region outlet temperatures generally increased somewhat more than would be expected from the power change.

Calculations done during and in support of various testing performed from 40% to 100% power and analyses of data from these tests indicate that, even before a redistribution, significant region outlet temperature measurement discrepancies exist in the seven NW boundary regions (Regions 20 and 32-37). Evaluations of observed differences between calculated and measured region peaking factors and steam generator module helium inlet temperatures have provided further evidence of region outlet temperature measurement discrepancies in the NW boundary regions. Thus, the region outlet temperature redistribution imposes additional perturbations of pre-existing measurement discrepancies in the NW boundary regions as well as real changes in the outlet temperatures of the remaining regions.

The region outlet temperature redistributions are the result of small in-core displacements. These displacements are similar in nature to the initial motion which occurred during fluctuations; however, they are not cyclic. These small (on the order of 0.10 in. or less) displacements cause changes in gap distribution, changes in crossflow, and (for the seven NW boundary regions, Regions 20 and 32-37) changes in the amount of cool transverse helium flow (Type II flow) along the sleeve(s) surrounding the region outlet temperature thermocouples. These observations are consistent with a general, although asymmetric, tightening of the inner regions of the core, where the gaps around the outer regions generally increase and gaps between inner regions generally decrease.

A method for operating the reactor has been developed, which accounts for region outlet temperature measurement discrepancies both before and after a redistribution. Under this operating method, the seven NW boundary regions, which are susceptible to outlet temperature measurement errors, will be operated by comparison regions in a manner similar to that employed in test procedure RT-500K. For the other 30 regions in the core, indicated changes in the region outlet temperature which occur during a region outlet temperature redistribution are real. These temperature changes can be accommodated and corrected as desired by orifice valve adjustments as are made routinely following load changes. Appropriate revisions to the Technical Specifications that support this operating method have been submitted for NRC approval.

2.2 Evaluation

The Fort St. Vrain tests to 100% clearly demonstrated that the region constraint devices (RCDs) were successful in inhibiting the oscillation or fluctuations. The staff and their consultants are in general agreement with the General Atomic/Public Service Company of Colorado analysis of the temperature redistribution scenario, which has been corroborated further by analyses of post-RCD scram events, during which the core reverts back to its relaxed or un-corseted state, both by GA and by ORNL.

Considering the structural load problems to be potentially the most serious, it should be stressed that inspection of elements that withstood the original, pre-RCD, fluctuations showed no signs of damage, and therefore the lesser burdens of occasional redistributions should be even less taxing.

We agree with the GA/PSC conclusion that the redistribution events should not require reevaluation of the FSV FSAR accident analysis and should not be cause to limit FSV operation to 70 percent of rated power.

3.0 Refueling Region Outlet Temperatures, Definition 2.21

3.1 Introduction

A definition of the individual refueling region outlet temperature is provided to account for the potential temperature measurement errors induced as a result of Type II flow. Type II errors in the NW boundary regions 20 and 32 - 37 cause the core average outlet temperature to be lower than the actual average. The new definition utilizes the following formula to determine the outlet temperature of each NW boundary region from the measured outlet temperature and the power and flow characteristics of its comparison region:

$$T_o = T_{in} + \Delta T_{cr}\left(\frac{RPF_i}{RPF_{cr}}\right)\left(\frac{flow_{cr}}{flow_i}\right) \quad \beta$$

where T_o = region outlet temperature for region being operated based on comparison region,

T_{in} = core inlet helium temperature,

ΔT_{cr} = measured comparison region temperature rise,

RPF_i = physics calculated region peaking factor (RPF) for region being operated based on comparison region,

RPF_{cr} = physics calculated RPF for comparison region,

$flow_{cr}$ = flow through comparison region, inferred from its inlet orifice valve position,

$flow_i$ = flow through region being operated based on comparison region, inferred from its own inlet orifice valve position, and

β = factor to account for relative number of fuel columns in comparison region and NW boundary region, with values given by the following table:

		Comparison Region Type	
		7-column	5-column
NW Boundary Region Type	7-column	1	7/5
	5-column	5/7	1

3.1 (Continued)

This formula determines the outlet temperature of each NW boundary region from the measured outlet temperature and the power and flow characteristics of the corresponding comparison regions. The formula also provides a correction such that if the measured region outlet temperature in the NW boundary regions is higher than that established by the formula, the measured value is assumed to be correct.

3.2 Evaluation

Use of the comparison regions in conjunction with the formula for individual refueling region outlet temperatures is expected to reduce the margin of error to a point where the temperatures of the individual refueling regions are acceptable and correct. The formula presented above includes the correction factors to account for the power, flow and temperature characteristics of the comparison regions when they are used in determining the individual refueling region outlet temperatures; therefore, the staff finds the Technical Specifications using the propose formula acceptable.

4.0 Comparison Regions, Definition 2.22

4.1 Introduction

The previous section, Fluctuation Testing, stated that Type II flow has a cooling effect on the measured region outlet temperatures. This Type II helium flow travels down the gaps between columns and enters the sleeves around the thermocouples used for measuring the region outlet temperatures of the NW boundary regions. Since this Type II flow consists of helium which is cooler than the helium flowing through the core, the NW boundary region outlet thermocouples "see" a temperature which results from the mixing of hot and cooler helium. Therefore the NW boundary region outlet temperatures are measuring a temperature which is lower than the actual values. Comparison regions have been suggested to compensate for this temperature discrepancy. These comparison regions are regions that have the same characteristics as the NW boundary regions but are located in the core where transverse Type II flow does not affect the temperatures.

The NW boundary region contains core regions composed of 5 and 7 columns. Comparison regions are picked from the SE segment of the core containing the same number of columns. The comparison regions are specifically picked from an opposite segment of the core so that the Type II flow cannot influence the region outlet temperature readouts.

4.2 Evaluation

The use of comparison regions when evaluating the region outlet temperatures should provide the operators a means of determining the correct temperature as opposed to a misleading reading due to the cooling effects of Type II flow. The staff review of comparison regions indicates that their use will provide an effective and sufficiently accurate means of providing fuel temperatures as well as refueling region outlet temperatures.

The staff opinion is, however, that including the equation used for refueling region outlet temperatures in the Technical Specifications may cause unnecessary confusion to the plant operators especially if all the data is not readily available. Therefore, the equation should be referred to in the procedures to be followed by the operators and the appropriate training should be provided in the use and meaning of the formula.

5.0 Core Inlet Orifice Valves, Limiting Conditions for Operation

5.1 Introduction

During fluctuation testing the use of comparison regions was shown to be a way of minimizing region outlet temperature measurement errors. Comparison regions typically have calculated power densities of magnitude and shape as a function of control rod configuration similar to those of the corresponding region having an outlet temperature measurement error. Knowing the relative power densities (calculated) of the region susceptible to outlet temperature measurement error and that of the corresponding comparison region, and knowing the orifice valve position of the comparison region, the region susceptible to measurement error can be orificed to have an acceptable outlet temperature. Thus, the core inlet orifice valves of Regions 20 and 32-37 can be adjusted based upon the characteristics of their respective comparison regions such that their actual outlet gas temperatures are within the mismatch limits of Technical Specification Figure 4.1.7-1. The limits in this figure are more conservative than those used to develop the Core Safety Limit, Specification SL 3.1, and those contained in Specification LCO 4.1.7 at the time test RT-500 was conducted. In addition, Figure 4.1.7-1 directly limits the maximum region outlet temperature to 1555°F, which is consistent with Table 3.6-1 of the FSAR. By requiring that the limits in Figure 4.1.7-1 be met, maximum fuel temperatures are kept within FSAR-stated values regardless of the power level or the amount of core bypass flow which may exist.

Operational flexibility is provided by the fact that more than one region can be used as a comparison region for each of Regions 20 and 32-37. Any one of the comparison regions may be selected for use within the range of shim bank configurations for which it is best suited. Regions affected similarly by changes in the regulating rod position (Region 1 control rod pair) are preferred as comparison regions. Whenever possible, it is ppreferred that paired regions be of the same type, i.e., 7-column regions compared with other 7-column regions and 5-column regions compared with other 5-column regions.

Use of comparison regions requires that conditions in the comparison regions (power, flow, and outlet temperature) be well known. Accordingly, LCO 4.1.7c includes a limit on the allowable percent region peaking factor (RPF) discrepancy in a comparison region. RPF discrepancies result from combinations of errors or uncertainties in measured region outlet temperature, region flow inferred from orifice valve position, and calculated region power. The regions to be selected as comparison regions are not susceptible to significant cooling due to Type II flow effects. Accordingly, the indicated region outlet temperatures for these regions are considered to be reliable within the $\pm$50°F range used to develop the Core Safety Limit, Specification SL 3.1. The major cause of excessive RPF discrepancies, if any exist in candidate comparison regions, is uncertainty in flow through the region inferred from inlet orifice valve position.

Under the comparison region method of operation, excessively negative RPF discrepancies in a comparison region could result in prolonged, high fuel temperatures in the NW boundary region being paired with the comparison region. A negative RPF discrepancy would be obtained in a comparison region if the flow through the region were larger than the flow one would infer from orifice valve position (e.g., due to cross flow entering the region). This additional flow would result in an actual lowering of the measured comparison region RPF which is, in fact, calculated from measured region outlet temperature and inferred region flow. If a region with an excessively negative RPF discrepancy were being used as a comparison region, one would infer that the outlet temperature of the NW boundary region, based upon comparison region outlet temperature, power, and flow characteristics, is lower than it may be in fact. However, due to the effects of Type II flow upon the region outlet temperature measurement in the NW boundary regions, the operator would have no directly measurable indication of this possible discrepancy. Therefore, LCO 4.1.7c limits the RPF discrepancy in a comparison region to minus 10 percent.

A flow-induced RPF discrepancy of minus 10% in a comparison region can cause a fuel temperature increase in a NW boundary region ranging from about 100°F to about 180°F over the nominal value. The size of the increase is a function of the RPF/intra-region power tilt combination in the NW region.

Regions with the highest time-averaged fuel temperatures are expected to experience the most fuel kernel migration during normal operation. The fuel subject to the maximum time at high temperature in the core will experience a maximum temperature of 2120°F and an end-of-life temperature of 2012°F. The time-averaged temperature of this fuel is about 2050°F. Less than 1% of the coated particles in the core experience these worst conditions. For a NW boundary region RPF/tilt combination necessary to maintain the fuel at a time-averaged temperature of approximately 2050°F, a minus 10% RPF discrepancy in the comparison region has been shown to result in an increase of 150°F in the NW boundary region fuel temperature. The fuel temperature in this small amount of fuel, therefore, may reach approximately 2200°F, a value below the 2372°F design maximum fuel temperature.

For a RPF/tilt combination necessary to produce a peak NW boundary region fuel temperature of 2300°F, the FSAR equilibrium cycle core maximum was assumed. For this RPF/tilt combination, a minus 10% RPF discrepancy in the comparison region would result in an increase of about 180°F in the NW boundary fuel temperature, resulting in a peak fuel temperature of about 2480°F. However, core physics analyses have consistently indicated that the large intra-region power tilts necessary to produce higher fuel temperatures (such as the 2300°F FSAR equilibrium cycle core maximum) do not persist for long periods of time and have indicated that such conditions (RPF/tilt combinations) are usually found only in interior regions and not in the NW boundary regions. This 2480°F peak fuel temperature is, nevertheless, below the 2732°F local short term peak fuel temperature limit in Section 3.2.3.3 of the FSAR.

5.2 Evaluation

In our review of LCO 4.1.7, we recommended that the method used for calculation of the average core outlet temperature account for the measurement errors in the suspect regions. PSC's proposed method does this, and its accuracy relative to an ideal flow-weighted average calculation was shown to be sufficient.

Also, the parenthetical explanatory expression in LCO 4.1.7c was changed to read:

> (i.e., RPF measured shall not be less than 90% of RPF calculated).

This was done to clarify the measuring and prevent any possible ambiguity.

We have reviewed the analyses presented by General Atomic Company and Public Service Company of Colorado and conclude that the proposed revision to the Technical Specification in all instances imposes more restrictive and conservative conditions on plant operation; in each step of the analysis PSC has introduced worst-case assumptions. Therefore, we conclude that a minus 10 percent limit in the RPF discrepancy should result in fuel temperatures which are, over the long term, well within the FSAR design maximum fuel temperature of 2372°F, and over the short term, well within the FSAR local short term peak fuel temperature limit of 2732°F.

6.0 Region Peaking Factor (RPF) Surveillance

6.1 Introduction

The region outlet temperature in each of the NW boundary regions is determined by an equation which uses a ratio of the physics calculated RPF in the NW boundary region to the RPF in a corresponding comparison region. The calculated and measured RPFs will change during a refueling cycle as fission product inventories saturate, fissile material and burnable poison are depleted, control rods are withdrawn from the core, and region flow characteristics change. Accordingly, to assure that the appropriate RPFs are used in determining the region outlet temperatures of Regions 20 and 32-37, and to assure that the limit on RPF discrepancy for comparison regions in LCO 4.1.7 is met, Surveillance Requirement SR 5.1.7, Region Peaking Factor Surveillance, has been established.

In order to characterize the variation in RPF ratios with burnup during a fuel cycle, calculations are performed using the GAUGE code. Data taken during cycle 3 fluctuation testing showed that RPF ratio may either increase or decrease with burnup. These variations in RPF ratio over the course of each surveillance interval can be predicted with the GAUGE code, and the calculations can be updated at each surveillance to reflect the actual operating history of the reactor over the preceding surveillance interval. Increases in RPF ratio with burnup can result in a nonconservative assessment of NW boundary region outlet temperatures if these increases are not periodically taken into account.

The frequency of calculated RPF surveillance specified in SR 5.1.7 requires that calculated RPFs be checked more often early in the refueling cycle (at beginning of cycle, at 20 ± 5 EFPD, and at 40 ± 5 EFPD), when analyses indicated more rapid changes in RPF ratios. The largest increase in RPF ratio which has been projected to occur during these intervals in 12%, which would have occurred between beginning of cycle (assumed to be 5 EFPD) and 25 EFPD during Cycle 3.

The frequency for surveillance of percent RPF discrepancy in Specification SR 5.1.7 has been established based upon conservative evaluations of potential fuel kernel migration. As stated in the basis of Technical Specification SL 3.1, the Core Safety Limit has been constructed to assure that a fuel kernel migrating at the highest rate in the core will penetrate a distance less than the combined thickness (i.e., 70 microns) of the buffer coating plus inner isotropic coating on the particle. It is further noted in the basis of SL 3.1, that the maximum fuel kernel migration expected for the fuel with the most damaging temperature history is less than 20 microns. Thus, out of a total inner coating thickness of 70 microns, only 50 microns were assumed to be available in establishing the limits in SL 3.1. To establish an appropriate frequency for RPF discrepancy surveillance, the time required for a fuel kernel to migrate 20 microns as a function of fuel temperature was evaluated, based on analytical data that showed migration of the ThC kernels to be more rapid than (Th/U)C kernels. Under SR 5.1.7 Surveillance of RPF discrepancy will be conducted monthly and the longest interval may be 45 days. The data indicated that in order for fuel kernels to migrate 20 microns over a 45 day period, the fuel must be exposed to a constant temperature of approximately 2530°F.

In order for a NW boundary region to reach temperatures in excess of 2530°F, the RPF discrepancy in its comparison region must be quite large and negative. If the RPF/tilt combination in the NW boundary region were such as to produce the expected maximum time-averaged peak fuel temperature of 2050°F, a comparison region RPF discrepancy of about minus 25% would be required to increase the fuel temperature to 2530°F.

A minus 10% RPF discrepancy, as allowed by LCO 4.1.7, could increase the peak fuel temperature by 150°F to approximately 2200°F. This is less than the FSAR design maximum fuel temperature of 2372°F and the 2530°F temperature at which a fuel kernel would migrate 20 microns over a 45 day maximum RPF discrepancy surveillance interval.

Another mechanism by which an RPF discrepancy might be imposed on a comparison region during a surveillance interval is by a region outlet temperature redistribution. If, as a result of redistributions, the flow through the comparison region were to increase (by opening of a crossflow (jaws) path), a negative RPF discrepancy would occur. A review of all the data for redistributions obtained during RT-500 testing indicated that the largest decrease in region outlet temperature (indicative of the opening of a crossflow path) relative to the expected temperature change which has occurred in a candidate comparison region is 90°F. Assuming an RPF/tilt combination necessary to produce the maximum expected time average fuel temperature (i.e., 2050°F) occurs in a NW boundary region, then, a decrease in the corresponding comparison region outlet temperature of 90°F, if allowed to go uncompensated, could result in a 150°F increase in the fuel temperature of the NW boundary region. Thus a peak fuel temperature of 2200°F could be obtained. Again, this temperature is below the FSAR design maximum fuel temperature of 2372°F. It is also below 2530°F, the temperature at which a fuel kernel would migrate 20 microns during a 45 day maximum RPF discrepancy surveillance interval.

6.2 Evaluation

The NRC staff and our consultants reviewed the analyses submitted in support of the proposed Technical Specifications and conclude that the surveillance frequency is such that appropriate corrective action can be taken in time to assure that fuel particle coating integrity is maintained. The analyses presented by GA and PSC shows that for any previously-observed redistribution, following RCD installation, the maximum temperature shift in a comparison region would be such that a 150°F maximum fuel temperature increase could result in a NW boundary region. However, the effects of multiple redistribution events and the possibilities of larger effects due to large core pressure drops could result in larger fuel temperature increases. Furthermore, a series of redistribution events could possibly result in temperature changes in the same direction and may not be amenable to the assumption of radomness. Also, the analysis relating to the fuel kernel migration effects considers a fuel temperature rise due to a single event while kernel migration is a cumulative effect. Because of the uncertainties in refueling region behavior following redistribution events, we require the following:

1) The RPF discrepancies be recalculated after each observed redistribution during the next shift that has the capability and facilities to calculate new RPF discrepancies.

2) As the FSV reactor approaches higher core pressure drops and the temperature redistribution phenomenon displays characteristics and conditions beyond those presently anticipated, these characteristics and conditions will be re-evaluated in terms of the proposed Technical Specifications and changes will be made appropriately.

7.0 Conclusions

In support of its request for release from the 70% power operation limitation for Fort St. Vrain with region constraint devices installed, the Public Service Company of Colorado (PSC) has submitted, by letter dated July 6, 1982, the following documents for staff review:

a) Response to ORNL Questions (ORNL Letter Dated March 15, 1982);

b) Final Report "Testing and Operations of Fort St. Vrain Up to 100% Power";

c) Proposed Technical Specifications regarding LCO 4.1.7 for Core Inlet Orifice Valves, and surveillance requirement 5.1.7 for Region Peaking Factor Surveillance; and

d) Safety Evaluation Report, Technical Specifications for Operation of Fort St. Vrain With Region Outlet Temperature Measurement Discrepancies.

These documents have been reviewed by both the staff and our technical consultants at Oak Ridge National Laboratory and the finding was made that operation of Fort St. Vrain at 100 percent power poses no undue risk to public health and safety.

However, the staff requests, as recommended by ORNL, that the following items be carried out by PSC:

a) Data-taking procedures similar to those used in the RT-500K test series shall be used as FSV operation is extended into higher core pressure drop (>5 psid) regimes. A description of the procedures when they are developed, along with the acquired data up to 100% power shall be submitted for staff information.

b) A description of how the core inlet helium temperature is calculated, what errors are expected and how these errors influence the computed outlet temperatures of regions 20 and 32-37, shall be incorporated and used in the procedures developed for plant operation up to 100% power.

c) The proposed technical specifications shall be revised as follows:

    1) The percent region peaking factor (RPF) discrepancy surveillance requirements (SR 5.1.7) shall include recalculations of the discrepancy after each region outlet temperature redistribution event by the next shift capable of doing so with proper engineering support.

    2) The equation for the calculation of the region outlet temperature for regions 20 and 32-37 shall be included in Technical Specification 2.21 or in the procedures to be followed.

The above items are procedural changes for clarification purposes only. They do not affect the conclusion that full power operation at FSV poses no undue risk to public health and safety. Therefore, the staff has concluded that the PSC request for release from 70% power operation limitation is acceptable. The Fort St. Vrain reactor may be operated at full 100 percent steady state power as approved in the staff's original findings and modified by the proposed Technical Specifications.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 5, 1982

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APPENDIX A

CHRONOLOGY OF FORT ST. VRAIN LICENSING ACTIONS

PERTAINING TO PLANT OPERATION, SAFETY EVALUATIONS AND LICENSE AMENDMENTS

DATE	TITLE
September 17, 1968	Commission issued a construction permit for the Fort St. Vrain Nuclear Generating Station.
November 4, 1969	Public Service Company of Colorado submitted the FSAR as amendment 14 to its application for a construction permit and license.
January 20, 1972	Safety Evaluation by the Division of Reactor Licensing, U. S. Atomic Energy Commission in the matter of Public Service Company of Colorado - Fort St. Vrain Nuclear Generating Station, Docket No. 50-267. This document pertained to the review of the Final Safety Analysis Report prior to issuance of an operating license.
June 12, 1973	Supplement No. 1, Safety Evaluation by the Directorate of Licensing, U. S. Atomic Energy Commission in the matter of Public Service Company of Colorado - Fort St. Vrain Nuclear Generating Station, Docket No. 50-267. This document pertained to postulated high energy pipe ruptures outside containment.
December 21, 1973	License No. DPR-34 issued for the operation of the Fort St. Vrain Nuclear Generating Station.
May 17, 1974	Safety Evaluation by the Directorate of Licensing Supporting Amendment No. 1 to License No. DPR-34. Changes the Technical Specifications by: (1) making exceptions to requirements for installation of secondary closures during certain initial low power physics testing, (2) revising specifications for monitoring during certain radioactive effluent releases, (3) revising specification for tendon load cell and PCRV concrete crack surveillance, (4) revising certain specifications for checks, calibrations, and testing of loop shutdown system, and (5) redefining certain administrative responsibilities and authorities of the offsite Nuclear Facility Safety Committee.
June 27, 1974	Safety Evaluation by the Directorate of Licensing Supporting Amendment No. 2 to License No. DPR-34. Changed the Technical Specifications to revise the organization of personnel for Fort St. Vrain Nuclear Generating Station.
July 12, 1974	Safety Evaluation by the Directorate of Licensing Supporting Amendment No. 3 to License No. DPR-34. Changed the Technical Specifications to allow low power reactor operation with a helium environment in the reactor during Phase I of the power ascension program.

Date	Title
November 11, 1974	Safety Evaluation by the Directorate of Licensing Supporting Amendment No. 4 to License No. DPR-34. Changed the Technical Specifications to permit revision of (1) radial power peaking factors under certain operating conditions and (2) the number of core regions allowed the maximum deviation in outlet temperature from the average core outlet temperature.
December 19, 1974	Safety Evaluation by the Directorate of Licensing Supporting Amendment No. 5 to License No. DPR-34. Changed the Technical Specifications to permit revised staffing requirements for plant operating shifts.
January 23, 1975	Safety Evaluation by the Division of Reactor Licensing, Supporting Amendment No. 6 to License No. DPR-34. Changed the Technical Specifications to permit a change in calibration frequency for one adjustment of the wide range power instrumentation and added a calibration requirement for the linear range power instrumentation.
April 17, 1975	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 7 to License No. DPR-34. Changed the Technical Specifications to permit bypass of the two-loop trouble scram when the reactor mode switch is in the "fuel loading" position.
December 1, 1975	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 8 to License No. DPR-34. Permitted the possession and use of additional radioactive sources for the purpose of calibration and instrument checks.
December 29, 1975	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 9 to License No. DPR-34. Changed the Technical Specification to permit a reduction in the helium circulator high-speed trip when operating on water-driven Pelton turbines.
January 27, 1976	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 10 to License No. DPR-34. Changed the Technical Specifications to permit a change in the procedures to be followed in the event of trouble with the hydraulic power system.

Date	Title
April 15, 1976	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 11 to License No. DPR-34. Changed the wording in the Technical Specifications to eliminate an inconsistency in the plant protection system labeling and the Final Safety Analysis Report.
April 26, 1976	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 12 to License No. DPR-34. Changed the Technical Specifications to add surveillance requirements for helium circulators and helium circulator Pelton wheels.
June 18, 1976	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 13 to License No. DPR-34. Changed the Technical Specifications to: (1) add requirements for operation of analytical system moisture monitors between reactor shutdown and 5 percent power; also calibration frequency for these monitors is stated; (2) revise allowable primary system impurity levels and method of specifying moisture impurity from parts per million to dew point temperature; (3) add a definition of operable dew point moisture monitors; (4) add functional checks and tests for dew point moisture monitors; (5) revise the core reactivity status surveillance and limiting conditions for operation; (6) isolate the helium storage system from the helium circulator buffer helium system when the reactor is in operation; (7) allow bypass of plant protective system moisture monitors for testing during the startup testing program; and (8) add reporting requirements.
June 18, 1976	Safety Evaluation by the Office of Nuclear Reactor Regulation supporting amendment no. 14 to licensee no. DPR-34. Revised the Technical Specifications to add requirements for: (1) backup pumping capability to the fire water system; (2) surveillance for the added pumps; and (3) an additional class 1E power source for the plant protective system.
June 24, 1976	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 15 to License No. DPR-34. Changed Technical Specifications to add requirements for operability and surveillance of shock suppressors.

Date	Title
November 17, 1976	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 16 to License No. DPR-34. Revised the section of the Technical Specifications relating to administrative controls.
December 8, 1976	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 17 to License No. DPR-34. Temporarily revised the provisions in the Technical Specifications relating to operation of the bearing water makeup pumps in the primary coolant system.
October 28, 1977	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 18 to License No. DPR-34. Permitted Stage 2 operation up to 70 percent of rated thermal power.
February 23, 1979	Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 19 to License No. DPR-34. Incorporates the Fort St. Vrain Amended Security Plan as part of the license.
April 20, 1979	Safety Evaluation by the Office of Nuclear Reactor Regulation supporting amendment no. 20 to license no. DPR-34. Revised the Technical Specifications to: (1) install eight test fuel elements into the reactor core at the first refueling, and (2) install PGX graphite surveillance specimens into five bottom transition reflector elements of the reactor core.
June 6, 1979	Safety Evaluation by the Office of Nuclear Reactor Regulation supporting amendment no. 21 to license no. DPR-34. Revised the Technical Specifications to: (1) modify the fire protection system for the three room complex, the Auxiliary Electric Room, the 480 Volt Switchgear Room and the congested cable areas; this constitutes Stage III fire protection implementation; (2) convert the Interim Alternate Cooling method to the final Alternate Cooling Method; (3) test the reactor building louver system on a quarterly basis; (4) eliminate the manual isolation of the high pressure helium supply from the helium circulator buffer supply header; and (5) add two firewater booster pumps to the firewater system to provide adequate capacity to operate a circulator water turbine and supply emergency cooling water for safe shutdown cooling.

Date	Title
August 19, 1980	Safety Evaluation by the Office of Nuclear Reactor Regulation supporting amendment no. 22 to license no. DPR-34. Revised the Technical Specifications to (1) change the amount of diesel fuel in each diesel generator set day tank to 325 gallons; (2) update company reorganization based on NRC requirements; (3) change the number of hours that the ACM diesel generator can operate with 10,000 gallons of fuel to 108 hours; (4) alter the Fire Protection Technical Specifications to follow the requirements of STS on Fire Protection; (5) change the frequency and method of Reactor Protective System Surveillance to satisfy the requirement of IEEE-279-1971; (6) update the listing of all snubbers; (7) change the fissile particle thorium to uranium ratio to reflect "as manufactured" specifications and (8) change the values for core region peaking factors and outlet temperature dispersions to reflect existing values in conjunction with accident reanalyses in support of full power operation.
March 16, 1981	Safety Evaluation by the Office of Nuclear Reactor Regulation supporting amendment no. 23 to license no. DPR-34. The amendment includes license conditions for the test program and revises the Technical Specifications to: (1) extend the minimum sample flow limits to cover the reactor power range of 70 to 100 percent, (2) define the times to start depressurization, (3) extend the core residence time of the fuel test elements and (4) specify operator action time limits for power-to-flow ratios as per S.L. 3.1.
November 9, 1981	Safety Evaluation by the Office of Nuclear Reactor Regulation supporting amendment no. 24 to license no. DPR-34. The amendment consists of two parts, each with a different effective date: (1) a temporary change to the Technical Specifications in response to a telecopied request of October 26, 1981 for relief from LCO 4.2.7.c PCRV Pressurization until November 27, 1981. This part of the amendment was authorized by telephone on October 27, 1981 and was confirmed by letter dated October 28, 1981; (2) in response to PSC application dated October 30, 1981 a change to the Technical Specifications section 7.1, Organization, Review and Audit-Administrative Controls, was made to incorporate the Shift Technical Advisor position and to reflect recent organizational changes.
March 2, 1982	Safety Evaluation by the Office of Nuclear Reactor Regulation supporting amendment no. 25 to license no. DPR-34. The amendment includes a modification of the license to permit possession of additional sources and revises the Technical Specifications to: (1) specify the period of

Date	Title
	time and conditions under which the Unit Auxiliary Transformer can be removed from service with the reactor at power, (2) substitute the requirement for an Annual Operating Report with an Annual Occupational Exposure Report and a Monthly Operation Report, (3) allow manual reset of the 30% bistables for operation at less than 30% but greater than 10% power, (4) require operability of snubbers when the reactor is at power, (5) establish an upper time limit for loss of voltage to 480 volt buses and the method to be used during surveillance testing, (6) substitute outdated requirements with those that comply to 10 CFR 20.103, (7) revise the method of performing thermocouple testing, (8) specify the testing of carbon sample cannisters as per NRC requirements, and (9) include two additional snubbers.
March 18, 1982	Safety Evaluation by the Office of Nuclear Reactor Regulation supporting amendment no. 26 to license no. DPR-34. The amendment revises the Technical Specifications to: (1) permit the interspace between primary and secondary closures of the steam generator modules to be maintained at a pressure slightly above cold rehear steam pressure; and (2) set a limit on the possible release of primary coolant activity through the primary closure seals of no greater than 1.4 curies per day.
August 5, 1982	Safety Evaluation by the Office of Nuclear Reactor Regulation supporting amendent no. 27 to license no. DPR-34. The amendment modifies the license to include a requirement to: (1) maintian a Safeguards Contingency Plan to be fully implemented, in accordance with 10 CFR 73.40(b), within 30 days of this approval; and (2) maintain a Guard Training and Qualification Plan, to be followed, in accordance with 10 CFR 73.55(b) within 60 days of this approval by the Commission. With regard to Item 2, all security personnel shall be qualified within 2 years of the approval.
October 5, 1982	Safety Evaluation by the Office of Nuclear Reactor Regulation supporting amendment no. 28 to license no. DRP-34. The amendment removes a license condition and revises the Technical Specifications to: (1) define individual refueling region outlet temperatures, (2) define a comparison region, (3) limit the maximum mismatch between region outlet temperature and core average outlet temperature, and (4) add a new surveillance requirement to assure that the limit on comparison region RPF discrepancy is met and the values of RPF used to determine the outlet temperature of the seven NW boundary regions are correct.