SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT 22 TO FACILITY
OPERATING LICENSE NO. DPR-34

OF

PUBLIC SERVICE COMPANY OF COLORADO
FORT ST. VRAIN NUCLEAR GENERATING STATION
DOCKET NO. 50-267

8009040 513

TABLE OF CONTENTS

1.0 INTRODUCTION

Fort St. Vrain, a 330-MWe high temperature gas cooled reactor (HTGR), was designed by the General Atomic Company and is being operated by the Public Service Company of Colorado near Platteville, Colorado. On October 28, 1977, the Nuclear Regulatory Commission authorized operation of the reactor up to 70 percent of rated thermal power. All of the power ascension tests have been completed up to 70% of thermal power, which was authorized by Amendment 18 dated October 28, 1977.

This amendment deals with various design and administrative modifications that Public Service Company of Colorado will perform to ensure increased reliability of operation, maintenance and safety of their Fort St. Vrain nuclear generating station. These modifications have been requested by means of letters and requests for changes to the Technical Specifications and include:

1. Changing the amount of diesel fuel in each diesel generator set day tank to 325 gallons.

2. Company reorganizations based on NRC requirements.

3. Changing the number of hours that the ACM diesel generator can operate with 10,000 gallons of fuel to 108 hours.

4. Completion of the Fire Protection Technical Specifications to follow the requirements of Standard Fire Protection Technical Specifications.

5. Changing the frequency and method of Reactor Protective System Surveillance to satisfy the requirement of IEEE-279-1971.

6. Changing the table listing all snubbers in the plant to reflect an updated status as a result of additions.

7. Changing the fissile particle thorium to uranium ratio to reflect "as manufactured" specifications.

8. Changing 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.

This last item is one of three that limit the operation of Fort St. Vrain to 70% power. The other two are Helium Depressurization and Moisture Injection tests; these items will be discussed in the next Safety Evaluation Report of Amendment 23.

The Fort St. Vrain reactor is described in the Final Safety Analysis Report submitted for our review in November 1969. The final Safety Analysis Report, as amended, formed a basis for our January 20, 1972 safety evaluation report and a first supplement, which was issued on June 12, 1973. The operating license, DPR-34 was issued on December 21, 1973. The operating license has been amended twenty-two times, including the amendment supported by this safety evaluation. The Final Safety Analysis Report and other early documentation continues to support our safety reviews, as augmented by the additional information and the operational reports referenced herein.

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 Specifications, Public Service of Colorado provides "Reportable Event" reports and "Unusual Event" reports on safety items relating to abnormal, unusual or unanticipated events that occur during the course of plant operations. In addition to the reports received from the licensee, our safety 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 experience and policies, information developed on gas cooled 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.

2.0 AUXILIARY ELECTRIC SYSTEM

Public Service Company of Colorado requested a change to the Fort St. Vrain Technical Specifications, Limiting Conditions for Operation of the Auxiliary Electric System in their letter dated March 7, 1979 (P-79056). Specifically, the Technical Specifications, among other requirements, state that the reactor shall not be operated at power unless both the diesel-generator sets are operable and 500 gallons of fuel exist in each day tank. Public Service Company of Colorado proposes to change the amount of fuel in each day tank to 325 gallons.

Discussions with the applicant and the NRC Resident Inspector revealed that the diesel generator set day tank level control system does not start refilling the tanks until the level reaches 350 gallons. Each day tank is of a 500 gallon capacity and when the diesel generator is operating, the tank level obviously decreases below the 500 gallon mark without being refilled. If operation of the diesel generator is stopped when the day tank level is between 500 and 350 gallons, the fuel oil pump is not started automatically and the day tank level will remain below the 500 gallon limit thereby requiring manual topping-off.

We have reviewed the proposed change and conclude that a 325 gallon supply of fuel oil in each day tank is sufficient to bring emergency electric supplies on line at load without interruption owing to inadequate fuel supply prior to transfer of fuel from the main storage tank. Therefore, the change is found to be acceptable.

3.0 ADMINISTRATIVE CONTROLS

Public Service Company of Colorado requested changes to the Fort St. Vrain Technical Specifications, Administrative Controls sections dealing with organization and procedures in their letters dated January 23, 1979 (P-79015) and January 11, 1980 (P-80003). These changes reflect a company reorganizations based on NRC requirements related to Fire Protection and other issues during the past year.

We have reviewed the proposed changes in light of the requirements presented in the Standard Technical Specifications dealing with Fire Protection and the requirements of Quality Control and find them acceptable, contingent on receipt of notification that the program is fully in effect.

4.0 DIESEL GENERATOR

Public Service Company of Colorado requested a change to Fort St. Vrain Technical Specifications, Limiting conditions for Operation of the ACM Diesel Generator in their letter dated August 29, 1979 (P-79164). Specifically, the Technical Specifications state that the 10,000 gallons of fuel for the diesel generator provides for one week of operation of the generator.

Public Service Company of Colorado performed a fuel oil consumption test for the ACM diesel generator and determined that the initially established time that the 10,000 gallons of fuel oil would provide for full ACM load used a fuel consumption rate which was more conservative than that determined by the actual test. The actual test established the time that the 10,000 gallons of fuel oil will provide the diesel generator with operation at a full load of 900 KW or 4 1/2 days (108 hours) and not the originally established one week.

We have reviewed the proposed change and determined that the 108 hours is still sufficient time to obtain additional fuel oil from off-site sources. We therefore, find the proposed change acceptable.

5.0 FIRE PROTECTION

In the NRC Safety Evaluation Report supporting Amendment 21, June 6, 1979, the history of a three-stage fire protection improvement program initiated in 1975 was described together with approval of stage II implementation. Stages I and II had been approved earlier on June 18, 1976 and October 28, 1977. One additional item outstanding was noted in the June 6 safety evaluation. This was the revision of existing plant fire protection Technical Specifications to apply to other safety-related plant areas consistent with the requirements of the Standard Technical Specifications dealing with Fire Protection.

By letter dated P-79170 dated August 13, 1979, the licensee proposed the following new or revised Technical Specifications:

1. Specification LCO 4.2.6 - Fire Water System/Fire Suppression Water System, Limiting Condition for Operation.

2. Specification SR 5.2.10 - Fire Water System/Fire Suppression Water System, Surveillance Requirement.

3. Specification SR 5.2.24 - Circulating Water Makeup System, Surveillance Requirement.

4. Specification SR 5.10.3 - Smoke Detectors and Alarm, Surveillance Requirement

5. Specification LCO 4.10.5 - Fixed Water Spray System, Limiting Condition for Operation.

6. Specification SR 5.10.6 - Fixed water Spray System, Surveillance Requirement.

7. Specification LCO 4.10.6 - Carbon Dioxide Fire Suppression System, Emergency Diesel Generator Rooms, Limiting Condition for Operation.

8. Specification SR 5.10.7 - Carbon Dioxide Fire Suppression System, Surveillance Requirement.

9. Specification LCO 4.10.7 - Fire Hose Stations, Limiting Condition for Operation.

10. Specification SR 5.10.8 - Fire Hose Stations, Surveillance Requirement.

11. Specification LCO 4.10.8 - Yard Fire Hydrants and Hydrant Hose Houses, Limiting Condition for Operation.

12. Specification SR 5.10.9 - Yard Fire Hydrants and Hydrant Hose Hoses, Surveillance Requirement.

13. Section Introduction 4.10 - Fire Suppression Systems - Limiting Conditions for Operation.

14. Section Introduction 5.10 - Fire Suppression Systems - Surveillance Requirement

These fire protection Technical Specifications were proposed in a format consistent with the format of the existing Fort St. Vrain Technical Specifications rather than a format conforming with Standard Technical Specifications. The licensees' Technical Specifications were reviewed with the aid of a step-by-step comparison with the Standard Technical Specifications, submitted by the licensee on September 28, 1979, and found to meet the intent of the Standard Technical Specifications. Our review concluded that the licensees' fire protection Technical Specifications are acceptable. This review closes out the last remaining open item in the fire protection program review of Fort St. Vrain.

It must be noted, however, that while Surveillance Requirements 5.2.10 - Fire Water System/Fire Suppression Water System and 5.2.24 - Circulating Water Makeup System are acceptable for fire protection purposes, they are not yet acceptable from the standpoint of meeting the new inservice inspection and testing requirements which the licensee is developing. In its letter of November 30, 1979, these two systems were identified by the licensee as Category II systems and as detailed their surveillance requirement will be substantially more detailed than now expressed in the Technical Specifications.

6.0 REACTOR PROTECTIVE SYSTEM SURVEILLANCE

Public Service Company of Colorado requested a change to the Fort St. Vrain Technical Specifications, Surveillance and Calibration Requirements for Instrumentation and Control System (SR 5.4.1) in their letter dated March 23, 1976 (P-76075). Specifically, Public Service Company of Colorado requested to change table 5.4-4, startup channel calibration frequency (Item 1c) 2 and method, and Table 5.4-1, startup channel test method (Item 3c).

The staff concludes that item 1c under "frequency" column of Table 5.4-4 is not consistent with the other portions of the Technical Specifications and should be changed as requested to provide consistency. The proposed change does not affect considerations related to the health and safety of the public.

The staff has discussed the requested changes to item 3c under "Method" column of Table 5.4-1 and item 1c under "Method" column of Table 5.4-4 of the Technical Specifications with the licensee. The staff and the licensee have agreed that the following sentence shall be added: "The internal test signal shall be checked and calibrated to assure that its output is in accordance with the design requirements. This shall be done after completing the external test signal procedure by checking the output indication when turning the internal test signal switch." The staff requires this addition and concludes that the requirements of IEEE-279-1971 are satisfied and that these changes are acceptable.

7.0 SHOCK SUPPRESSORS, SNUBBERS

The Fort St. Vrain Technical Specifications dealing with the Limiting Conditions for Operation of Shock Suppressors or Snubbers state (Section 4.3.10e) that shock suppressors may be added to Class I systems without prior License Amendment to Table 4.3.10-1, provided a revision to Table 4.3.10-1 is included with a subsequent License Amendment request. The Public Service Company of Colorado elected to update Table 4.3.10-1 listing all Class I snubbers as they are added as a result of refined seismic analyses of their piping systems. As recommended by IE Bulletin 79-14, PSCo has reviewed their "safety-related" piping systems to verify that the deisgn drawings correspond to the "as-built" configurations. As a result of this review and additional requirements, PSCo has started a reanalysis of their piping systems and additional snubbers will be added or deleted as the analysis dictates.

8.0 DESIGN FEATURES

The Public Service Company of Colorado requested a change to the Fort St. Vrain Technical Specifications, Reactor Core Design Features, in their letter dated January 11, 1980 (P-80003).

During fuel fabrication in 1971, the coatings on the fissile particles were manufactured slightly thicker than specified and a problem arose in squeezing enough particles into a fuel rod to yield a thorium to uranium ratio of 4.25 to 1.

8.1 Background

On September 14, 1971, prior to final AEC approval of the then-proposed Fort St. Vrain (FSV) Technical Specifications, the FSV fuel design specification was modified to allow a decrease in the nominal Th:U ratio of fissile fuel kernels from 4.25:1 ($\pm$0.5) to 3.6:1 (+1.2, -0.2). However, no action was taken to modify the Th:U ratio described in Technical Specification D.F.6.1, which contains a description of the reactor core design features, including those of the fissile and fertile fuel particles. In the meantime, fuel with the 3.6:1 nominal Th:U ratio was used in the latter stages of FSV initial core production as well as in reload segments 7 and 8, and it is intended for use in future segments containing $(Th/U)C_2$ fissile fuel. The major impact of the decrease in Th:U ratio is a slight increase in the fissile kernel peak burnup for six-year-old fuel in FSV. The peak burnup for 4.25:1 fuel is expected to be 20% FIMA (fissions of initial metal atoms) whereas the peak burnup for 3.6:1 fuel is expected to be 22.4% FIMA. The overall ratio of Th:U in the core is unaffected by the change in the fissile particle specification.

8.2 Summary of Regulatory Evaluation

The major part of the technological basis for permitting the proposed technical specification change was provided in the attachment to the February 8, 1980 letter from Don Waremburg (PSC) to William P. Gammill (NRC). In that attachment, fuel performance under normal and design basis accident conditions was discussed in terms of the potential effects of the Th:U ratio change on fuel failure phenomena. In each case, test results were used to support the assertion that there were no discernable effects.

8.3 Fuel Performance

8.3.1 Normal Reactor Operation

For normal operation, three phenomena were considered: kernel migration, fabrication defects, and pressure vessel failure. In the case of kernel migration, coating failure is assumed to occur if a migrating kernel contacts the structural layers of a TRISO coating. Since the coating dimensions are the same for 4.24:1 and 3.6:1 kernels, this phenomena could be affected only if the change in Th:U ratio were to cause a change in the rate of kernel migration. However, test results indicate that the rate of migration is unaffected over a range of Th:U ratios from 4.25:1 to 1.60:1. The change in Th:U ratio from 4.25:1 to 3.6:1 should not, therefore, have any impact on coating failure associated with kernel migration.

In the case of the second coating failure phenomenon, viz., missing or defective coatings, the small decrease in Th:U ratio does not result in a change in specified coating properties or coating pressures. Therefore, the number of particles manufactured with missing or defective coatings would not change, and coating failure and fission product release resulting from this phenomena would not be affected.

The third failure phenomenon, called "pressure vessel" failure, is a function of kernel burnup, coating design, and operating temperature. From a summary of irradiation test results provided with the attachment to the February 8, 1980 letter from PSC, it was observed that the failure probability for fuel irradiated to burnups 27% FIMA was the same (0.004) as fuel irradiated to 20% FIMA. Hence, the irradiation test results indicate that the small increase in burnup (from a previous maximum of 20% to the current 22.4% FIMA) should have a negligible effect on pressure vessel failure.

8.3.2 Design Basis Accident Conditions

The loss of forced circulation (LOFC) is the most severe accident, with regard to fuel performance, analyzed in the FSV FSAR. Test results show that the major failure mechanisms include SiC-actinide metal reactions, SiC-fission product reactions, SiC decomposition.

In the FSV FSAR analysis of fuel performance during the LOFC accident, it was assumed that the fuel failure fraction was 0.05 for temperatures less than 1725°C and 1.0 for temperatures greater than or equal to 1725°C. More recent (1977) data, however, obtained on irradiated particles subjected to a heat-up that was conducted out-of-pile, show that fuel failure should not become significant until temperatures exceed 2100°C. Although that test was conducted on particles having 18.2% FIMA burnup, the effect of a slightly higher burnup (22.4% FIMA) should be negligible relative to the FSAR assumptions. As discussed in the attachment to the May 28, 1980 letter from PSC, heat-up tests of $(Th/U)C_2$ and UC_2 TRISO particles have shown that the failure rate is relatively insensitive to burnup. We, therefore, conclude that decreasing the Th:U ratio from 4.25:1 to 3.6:1 should not

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

result in significant increases in fuel failure and fission produce release during the worst case DBA.

8.4 Regulatory Position

We have completed our review of the documentation provided in support of the request for a change to Fort St. Vrain Technical Specification D.F.6.1. Based on our evaluation of the information supplied, we conclude that there is reasonable assurance that a decrease in the nominal Th:U ratio from 4.25:1 normal operation and design basis accident conditions and that the technical specification change is, therefore, acceptable.

9.0 Accident Reanalysis

9.1.0 Background

On November 1, 1977, the Public Service Company of Colorado submitted analyses in support of operation of the Fort St. Vrain plant at 100% of design power.

The power level of the Fort St. Vrain plant was originally limited to 70% of design power because of limitations in the helium purification system which must be used for depressurization in the event of a loss of forced circulation accident. These limitations were discovered during the review of the Alternate Cooling Method provided subsequent to the Brown's Ferry Fire and are addressed elsewhere. In addition, a separate problem arose in that tests disclosed that firewater delivery to the circulator Pelton wheels and steam generators was insufficient to keep predicted temperatures at or below those originally reported in the FSAR. PSCO justified, through analysis, that at a power level of 70% of design power, temperature predictions would fall at or below the original FSAR values. It was during these reanalyses that discrepancies between the values for core region peaking factors and outlet temperature dispersion used in the FSAR safety analyses and the values used in the plant technical specifications (which were higher) were identified. Accident reanalyses using the more limiting initial operating conditions permitted by the technical specifications were then submitted in support of proposed full power operation for Ft. St. Vrain. Additionally, the reanalyses for cores after initial refueling included the effects of pressure booster pumps which have since been installed in the firewater feedlines to the circulator pelton wheels. This modification was required to provide sufficient circulator flow to maintain acceptable fuel temperatures for the firewater cooldown accident case with the reactor at full power.

The evaluation which follows addresses the accident reanalyses (Nov. 1977 submittal) in support of full power operation.

9.2.0 Licensee Analyses

9.2.1 Scope

The licensee has submitted analyses of three accidents which are considered to be the most limiting. These are (1) Cooldown on one firewater-driven pelton wheel, (2) Rapid Depressurization/Blowdown, and (3) Permanent Loss of Forced Circulation. All of these reanalyses were performed with the RECA3 code, which was not used for the original FSAR analyses.

Differences in Technical Specification Peaking factors and Outlet Temperature Dispersion factors from those used for the original FSAR analyses are summarized in Table 2.1 below:

Table 2.1

	Peaking Factor	Outlet Temp. Dispersion
Original FSAR	1.78	54°F
Technical Specification	1.83	250°F

In support of the three bounding accidents identified, the applicant submitted the results of a review performed for all accidents originally analyzed in the FSAR. For those accidents affected by either Region Peaking factor or outlet temperature dispersion, a set of enveloping accidents was identified. The affected accidents are the Rod Withdrawal accident, the orifice closure accident, and steam in-leakage events. For the orifice closure accident, the conclusion that the original FSAR analyses were bounding was based on new data which showed that the fully closed orifice valve loss coefficient was approximately 1/2 of the value used for the FSAR.

The staff has reviewed the enveloping logic and the results of the review and finds acceptable the conclusion drawn by the applicant that the three accidents identified are bounding.

9.2.2 Analysis Methods

All reanalyses were performed using the RECA3 code. This code was not used to perform any previous analyses submitted to the NRC (i.e., for the FSAR). While the staff has not reviewed the code for applicability on a generic basis, we have determined the code to be acceptable for the specific analyses performed for the Fort St. Vrain Plant (See Section 3.0).

The applicant has also used the TAP and RATSAM codes to predict the core helium inlet temperature versus time and the system pressure versus time respectively, for input to the RECA3 analyses. Comparisons of these code predictions to alternate calculational methods, as well as the sensitivity of analysis results to uncertainties in these parameters are also provided in Section 3.0.

For the three accidents analyzed, the plant was assumed to be operating at 105 percent of full power, and 105 percent of full flow, with an initial power to flow ratio of 1.0. The applicant has stated that actual power to flow ratios may be in excess of 1.0 at indicated full power as indicated in Table 2.2. In addition, plant technical specifications permit operation at power-to-flow ratios in excess of 1.05 at power levels below 100% (see Figure 2.1). Maximum temperatures however, would occur for the 105 percent power level case due to the increased decay heat generation. This was confirmed by independent calculations by ORNL at selected points along the power-to-flow operating limit curve (Figure 2.1) as described in Section 3.0.

Table 2.2

	Indicated	Actual (worst case)	FSAR Assumption
Power	100%	102%	105%
Flow	95%	93.5%	105%
P/f ratio	1.05	1.09	1.0

9.2.3 Acceptance Criteria

The thermal limits for acceptable response of fuel and structures to postulated accidents are those originally approved in the FSAR and are provided in Table 2.3.

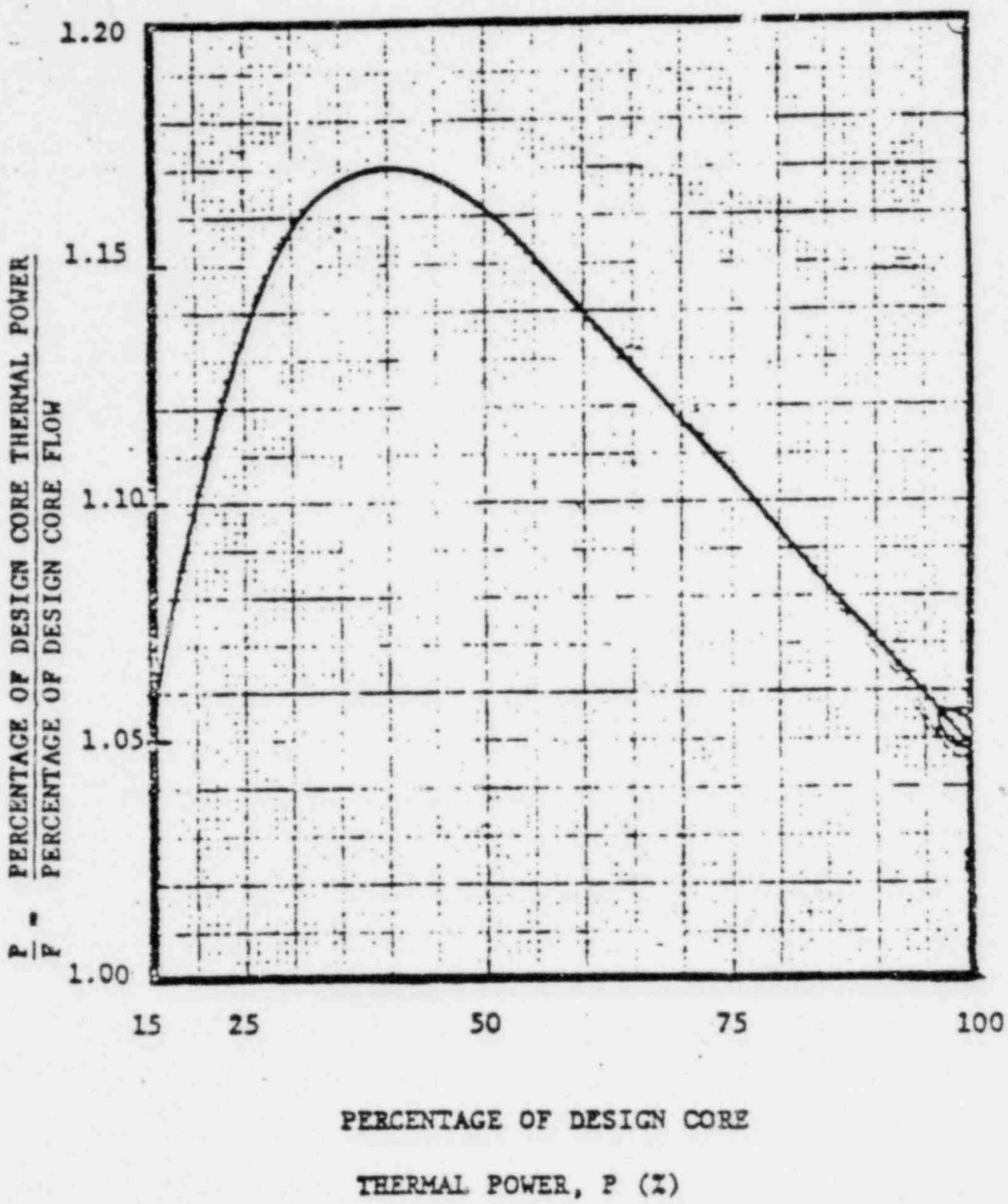


FIGURE 2.1

POWER-TO-FLOW RATIO TECHNICAL SPECIFICATION

CURVE FOR FORT ST. VRAIN

Table 2.3

Item	Temperature Limit
Fuel	2900°F
Steam Generator Inlet Ducts and Liners	2000°F
Upper Plenum Insulation and Cover Plates	1500°F

These limits do not represent points at which physical damage of the fuel or structure will occur, but rather are temperatures above which degradation is expected to increase significantly.

9.2.4 Analysis Results

The results of the RECA 3 reanalyses are compared to the temperature limits of Table 2.3 in Table 2.4.

9.3.0 Staff Evaluation

9.3.1 Methods Review

The staff has determined the acceptability of the applicant's analysis methods by (1) evaluation of key input assumptions to which the output is sensitive, (2) comparison of the results of applicable plant transient temperature data to temperature predictions for those transients using the RECA3 code,

Table 2.4

Event		Limit	RECA 3 Prediction
0-delay Firewater Cooldown/Initial Core	Fuel	2900°F	<2600°F
	Steam Generator Inlet Ducts & liners	2000°F	~1600°F
	Upper Plenum Insulation and Cover Plates	1500°F	<<1500°F*
Rapid Depressurization Blowdown	Fuel	2900°F	~2600°F
	Steam Generator Inlet Ducts & Liners	2000°F	1760°F
	Upper Plenum Insulation and Cover Plates	1500°F	<<1500°F
Permanent Loss of Circulation (LOFC)	Fuel	2900°F	<2900°F
	Steam Generator Inlet Ducts & Liners	2000°F	<2000°F
	Upper Plenum Insulation and Cover Plates	1500°F	~1500°F**

*Calculated temperatures were not reported by the applicant, since forced circulation is not lost and core inlet temperatures will remain close to the feedwater temperature.

**The top head liner temperature is calculated to not exceed the 1500°F limit provided the system is depressurized within 2 hours after LOFC.

(3) comparison of temperatures predicted by RECA3 to temperatures predicted by ORECA, and (4) comparison of analysis code predictions to hand calculations.

The ORECA code, which predicts the transient behavior of gas-cooled reactors, is similar in function to the applicant's RECA3 code. ORECA was developed by ORNL for the NRC.

The plant data used for code verification were from three reactor trips which occurred from power, and from one event in which all forced circulation was lost for approximately ten minutes.

9.3.1.1 Input Assumptions

As discussed in Section 2.2, the power-to-flow ratio used for all of the reanalyses was 1.05 and was confirmed by ORECA analyses to be the most limiting value. The results of these analyses are provided in Table 3.1.

In our review of initial conditions with respect to the allowable power-to-flow ratios in the technical specifications, it was noted that for limited periods of time, the technical specifications allow full power operation at power-to-flow ratios greater than 1.05 based on steady state time-at-temperature limits for fuel damage. The licensee considers operation in this region to be a degraded plant condition and has stated that normal practice is not to operate with power-to-flow ratios greater than 1.05. Since operation in this degraded mode has not been considered in the accident reanalyses, deliberate operation at power-to-flow ratios in excess of the curve shown as Figure 2.1 (Figure 3.1-1 of the technical specifications) is not acceptable to the staff. If the power-to-flow ratio limits of Figure 2.1 are exceeded, we will require that the operator act promptly to bring the plant within

TABLE 3.1

Results of ORNL Confirmatory Calculations Power-to-Flow Ratio Technical Specification

Case #	Initial P/F	$T_{core\ inlet}$ (°F)	Power %	Power Mwt	Flow %	Flow lbm/min	DBDA Max. Fuel	DBDA Max. Gas	LOFC Max. Fuel
1	1.05	768.3	100	842	95.2	54,760	2617	2313	2808
2	1.095	737.6	80	673.6	73.1	41,871	2335	2106	2555
3	1.14	706.9	60	505.2	52.6	30,084	2094	1928	2317
4	1.17	686.4	40	336.8	34.2	19,511	1923	1923	2072
5	1.10	734.2	20	168.4	18.2	10,418	1826	1736	1840
6*	1.091	740.4	102	858.8	93.5	53,601	2627	2322	2826
7**	1.043	773.0	104.3	878.3	100	56,500	2676	2351	2858

*Worst-case operational conditions

**Reference Case

allowable limits. We will require the licensee to propose technical specification revisions to conform to this position prior to approval of 100 percent power operation.

ORECA analyses were also performed to investigate the sensitivity of calculated coolant temperatures to variations in some of the input parameters. Results are listed in Table 3.2.

TABLE 3.2

	Peak TGas Out, °F	Hot Streak Temp., °F
Reference Case	2269	1927
Helium Flow (-20%)	2348	1995
Coolant Friction Factors (Laminar & Transition)(+20%)	2275	1926
Effective Coolant Heat Transfer Coefficent (-20%)	2252	1917
Afterheat (+20%)	2433	2034

From these studies, the most sensitive parameters were determined to be the helium flow through the core and the decay heat rate. The decay heat rate used for the RECA 3 analyses is the same as the decay heat rate curve approved in the FSAR.

The bypass fraction assumed for the accident reanalyses was 7.5% of the circulator flow*. There is some uncertainty in bypass flow because of the inability to directly measure flow, as well as the inability to measure flow path resistances and therefore determine relative flow splits. Calculation of the apparent bypass flow using the RECA 3 model indicated good agreement with the initial estimate. For the four scram tests to date, bypass fractions of 0.076 (40% power), 0.076 (50% power), 0.070 (60% power), and 0.063 (70% power) were calculated, which are in good agreement with the value of 7.5% assumed for the safety analyses. Moreover, analyses using ORECA indicate that even for bypass flow uncertainties upward of 20%, steam generator inlet temperature limits will not be exceeded. Based on the above, we find the use of a bypass fraction of 7.5% in the RECA analyses acceptable.

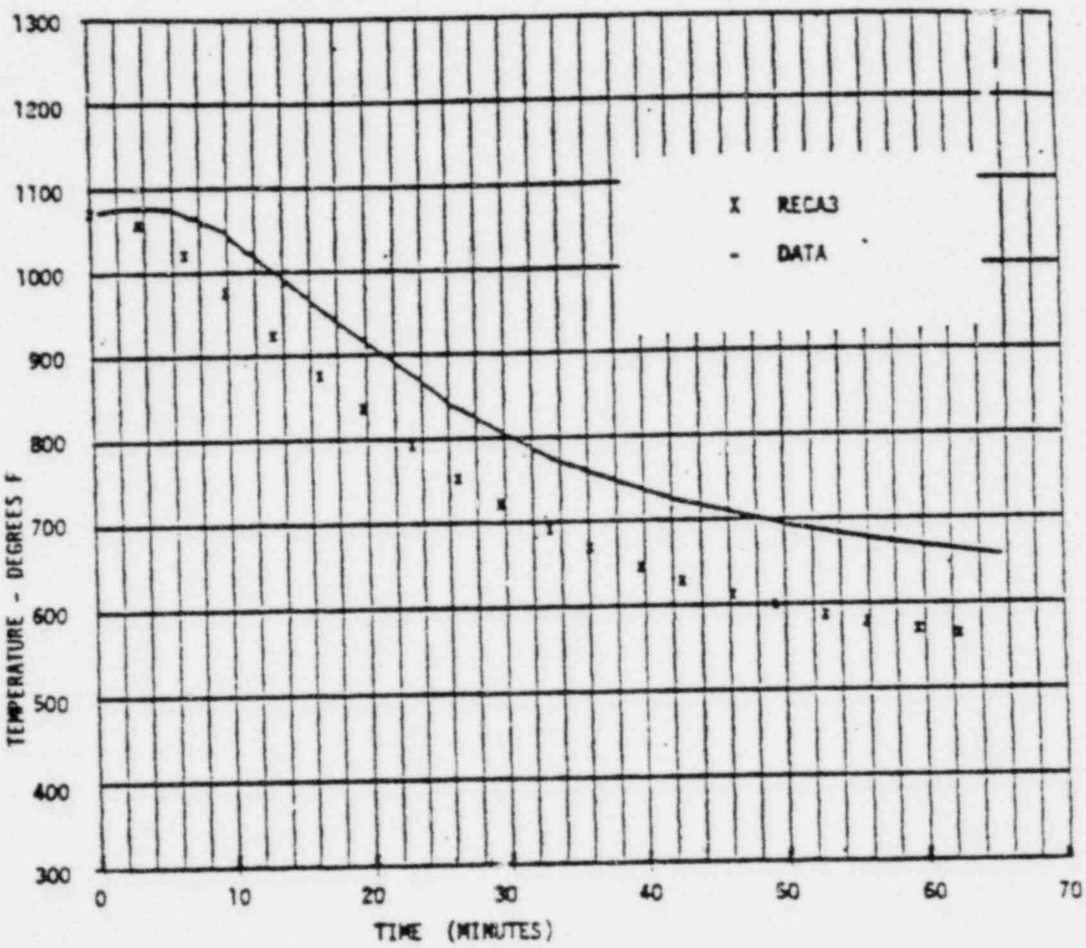
9.3.1.2 Code Verification

9.3.1.2.1 RECA3

The RECA3 comparisons to available scram data indicate that predictions of helium temperature in the maximum peaking factor refueling regions are in good agreement with the measured temperatures. However, the code underpredicted helium temperatures in the north-west quadrant of the core by as much as 50°F to 100°F in the 40-70 second time frame as shown in Figure 3.1. This discrepancy may be due to excess bypass flow through fuel region gaps in this

---

*The design bypass flow, or that flow which does not enter the core barrel, is 2.9 percent. The RECA analyses assume part of the unheated core flow as bypass. For these analyses, the bypass was input as 7.5 percent.



FORT ST. VRAIN SCRAM FROM 39 PERCENT POWER
ON 10/25/77 REGION 35

FIGURE 3.1

quadrant. Such observations are consistent with region outlet temperature fluctuation phenomena observed during plant operation. The fluctuations were most prominent in this region, and are believed to be due to the opening and closing of axial gaps between fuel blocks.

The discrepancy between the predicted and measured region outlet temperatures is of concern to the staff. We will therefore require that the applicant perform at least one verification transient subsequent to corrective action taken to eliminate the core fluctuations. This transient can be a reactor trip from power, and the verification should consist of comparisons of measured to predicted region outlet temperatures. Acceptable* predictions of the measured data, including resolution of the previously observed northwest quadrant discrepancies, will be required before full power operation is allowed. Alternatively, the licensee should identify an acceptable operating power level, based on accident analyses in which this uncertainty has been properly accounted for.

Comparisons of ORECA predictions to the plant trip data showed good agreement between the calculated and measured region outlet helium temperatures. In addition to the comparisons made to plant data, peak helium temperatures were also predicted for two of the three bounding accidents; the Design Basis Depressurization Accident (DBDA) and the Firewater Cooldown Accident (FWCD) with a zero time-delay assumed for the initiation of firewater cooling. For

*The predictive uncertainty should not be abnormally excessive for any refueling region when compared to the average.

the firewater cooldown accident, predictions were made for two core loadings, equilibrium, and initial. The results of these calculations compared to the applicant's predictions are given in Table 3.3.

TABLE 3.3

Event		RECA3 Prediction	ORECA Prediction
0 - Delay Firewater Cooldown/equilibrium Core	Peak average gas outlet temperature from core	1525°F	1509°F
	Peak gas outlet temperature for maximum region	1900°F	1873°F
0 - Delay Firewater Cooldown/initial core	Peak average gas outlet temperature from core	1500°F	1479°F
	Peak gas outlet temperature for maximum region	1900°F	1901°F
Design Basis Depressurization Accident/equilibrium core	Peak average gas outlet temp. from core	1700°F	1724°F
	Peak gas outlet temp. for maximum region	2350°F	2269°F
	Peak fuel temperature	2600°F	2557°F

9.3.1.2.2 RATSAM Code

The RATSAM code is used to predict system pressure versus time as input to the RECA3 calculations.

In order to assess the effect of uncertainties of the calculated pressure on the RECA3 calculated temperatures, the applicant performed both hand calculations of the transient pressure as well as RECA 3 reanalyses using a constant helium pressure of 700 psia.

The hand calculations of the transient pressure showed agreement with the general trends of the RATSAM-calculated pressure during the first hour after accident initiation. However, the RATSAM-calculated pressure was shown to slightly increase after one hour whereas the hand-calculated preessure continued to decrease beyond one hour.

To show that the effect of calculated pressure uncertainties did not have a large effect on the results, the applicant performed reanalyses with RECA3 assuming a constant 700 psia system pressure. These reanalyses were for the two accident analyses which require RATSAM input; the first 2 hours of the LOFC (prior to initiation of depressurization) and the first 1-1/2 hour delay of firewater to the pelton wheels.

The main result of the LOFC reanalysis assuming constant system pressure was that the time for the top head thermal barrier average cover plate temperature to reach 1500°F was reduced from 25 to 24 hours. The analysis also showed that the top head liner remained intact for both cases beyond 30 hours.

The results of the reanalysis of the FWCD with a 1-1/2 hour delay assuming constant system pressure showed that some temperatures in the core were reduced by 10 to 30°F from the case where system was calculated by the RATSAM code. The peak fuel temperature was also reduced by 53°F for this case. However, the average upper plenum temperature was shown to increase 138°F (to 1350°F) and the average PCRV top head thermal barrier cover plate temperature increased by 113°F (to 1152°F) at 1-1/2 hours. In neither case were the temperature limits of Table 2.3 exceeded.

The applicant has demonstrated that the general trend of initial pressure reduction predicted by the code is supported by hand calculations, and that with the assumption of a constant 700 psia system pressure (approximately 100 psia greater than the RATSAM predicted pressure), temperatures in the core, fuel and structures did not change the results significantly. Based on the above, the staff finds the use of RATSAM code acceptable for the purpose of predicting system pressures for the two FSV accidents analyzed.

9.3.1.2.3 TAP Code

The TAP code is used to calculate the temperature of the helium exiting the steam generators and entering the upper plenum and core as input to the RECA3 calculations. Although the TAP code has not been verified against data, hand calculations were performed by the licensee to confirm the calculational accuracy of the TAP code. These comparisons are provided in Figure 3.2 and show the TAP calculations to be in good agreement with the hand-calculated values of helium inlet temperature.

Moreover, the applicant stated that because of the excessive heat transfer capability of the steam generators at decay heat levels, coupled with the low core flow assumed subsequent to accident initiation, the helium temperature at

COMPARISON OF TAP AND HAND CALCULATED FORT ST. VRAIN CORE
INLET HELIUM TEMPERATURE UNDER FIREWATER COOLDOWN CONDITIONS

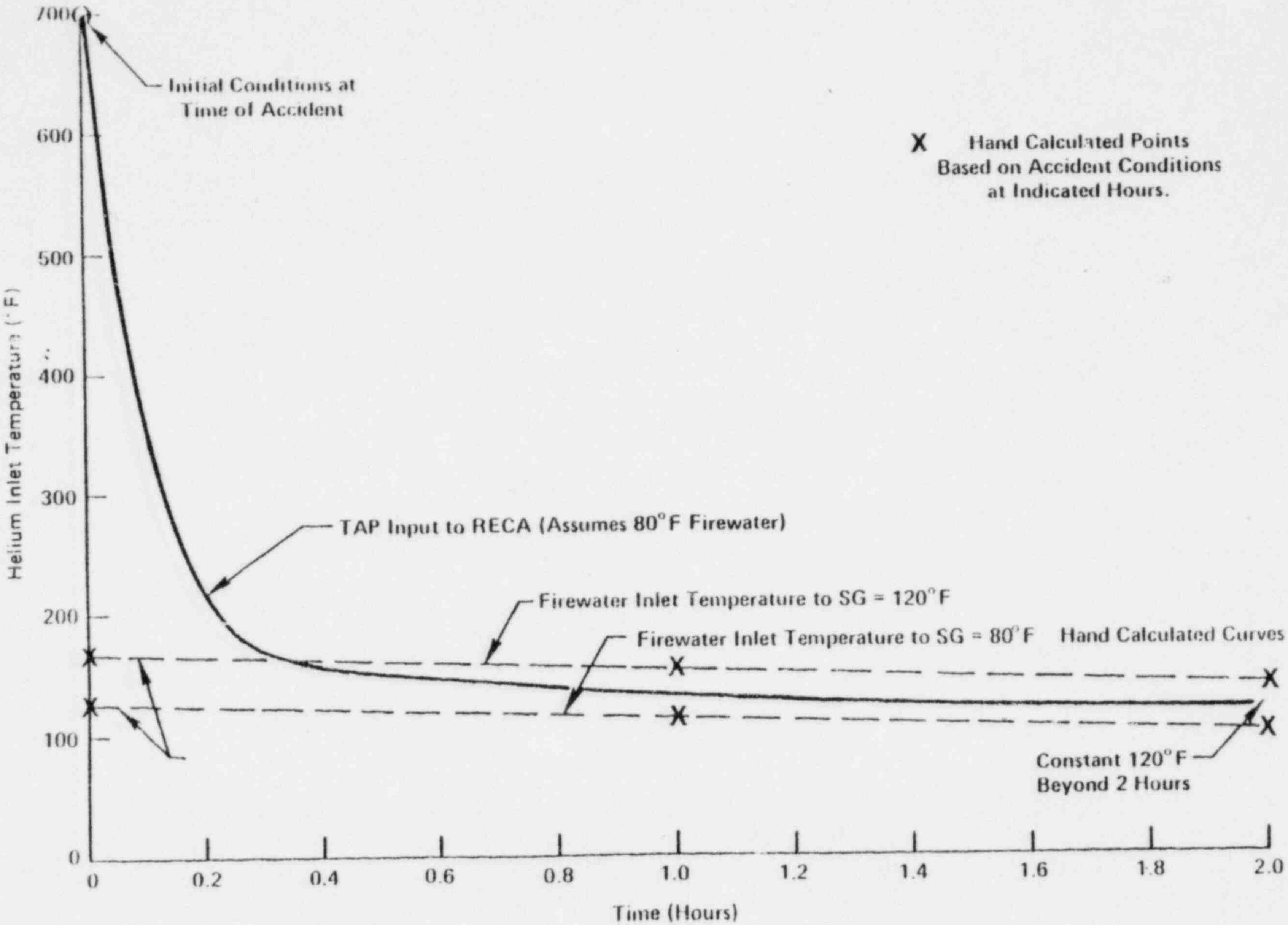


FIGURE 3.2

the exit to the steam generators will be approximately the same as the feedwater temperature, and therefore should not be a highly sensitive parameter. Based on the above considerations and the confirmatory hand calculations, the staff finds the use of the TAP code acceptable for the purpose of calculating the helium temperature exiting the steam generator.

9.4.0 Summary

The staff has reviewed the accident reanalysis submitted by the licensee in support of operation of the Fort St. Vrain plant at 100 percent of design power. Based on our review, we have concluded that the reanalyses provided are acceptable to justify full power operation. However, prior to operating at any power level above the present 70 percent restriction, the licensee must perform the following:

1. Provide for staff review and approval a minimum of one additional RECA3 code verification analysis of plant transient response. The transient response used for verification must be performed subsequent to corrective actions taken to eliminate the core fluctuations. Alternatively, an acceptable power level should be proposed which is based on accident analyses which account for this prediction uncertainty.

2. The licensee must propose, for staff approval, revisions to the plant technical specifications which will specifically preclude operation at power-to-flow ratios in excess of those for which the plant transient response has been shown to be acceptable.

10. Conclusions

Based on our review of the documentation referenced in this report, an evaluation of plant operations thusfar, evaluations of the plant through site visits by NRC technical specialists, and favorable reports by the NRC Office of Inspection and Enforcement, we conclude that the Technical Specifications can be revised as requested. This safety report describes the basis for this conclusion, and notes the conditions which apply.

Since the Fort St. Vrain reactor is the first and only plant of this size and type, and since a substantial base of experience comparable to that for light water reactors does not exist, the performance of the Fort St. Vrain reactor continues to be closely monitored by the NRC staff.

As related in Amendment No. 18, three items form a formal hold to operation of the reactor above 70% power: (1) Depressurization, (2) Moisture Monitors, and (3) Accident Reanalysis. This third item has been reviewed and our comments and conclusions presented in Section 9 of this Safety Evaluation Report. Items 1 and 2 are still under NRC review and should be resolved in the near future. Investigations into the power/temperature fluctuations continue at both the NRC and PSCo. Public Service Company of Colorado plans to perform acceptance tests of the installed Region Restraint Devices (Luci Locks) sometime in the fall of 1980 to demonstrate their ability to eliminate the fluctuations. The licensee plans to modify the Buffer Heluim System to provide two separate heluim lines, one for each loop. Is is anticipated that this split will eliminate most of the buffer-mid-buffer upsets experimenced at the Fort St. Vrain reactor. The physical split of the loops is planned for the next refueling outage, tentatively scheduled for May 1981.

These planned activities do not detract from the conclusion that the plant may be operated at 70% of rated power.

Finally, the NRC staff requires and expects the licensee to proceed expeditiously to resolve those matters as described in this safety evaluation report, and to also expeditiously complete all work required for 100% power operations.