SAFETY EVALUATION REPORT

FOR THE

FORT ST. VRAIN PLANT

ACCIDENT REANALYSES

IN SUPPORT OF

100% POWER OPERATION

8006200629

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.

2.0 Licensee Analyses

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.

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

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.

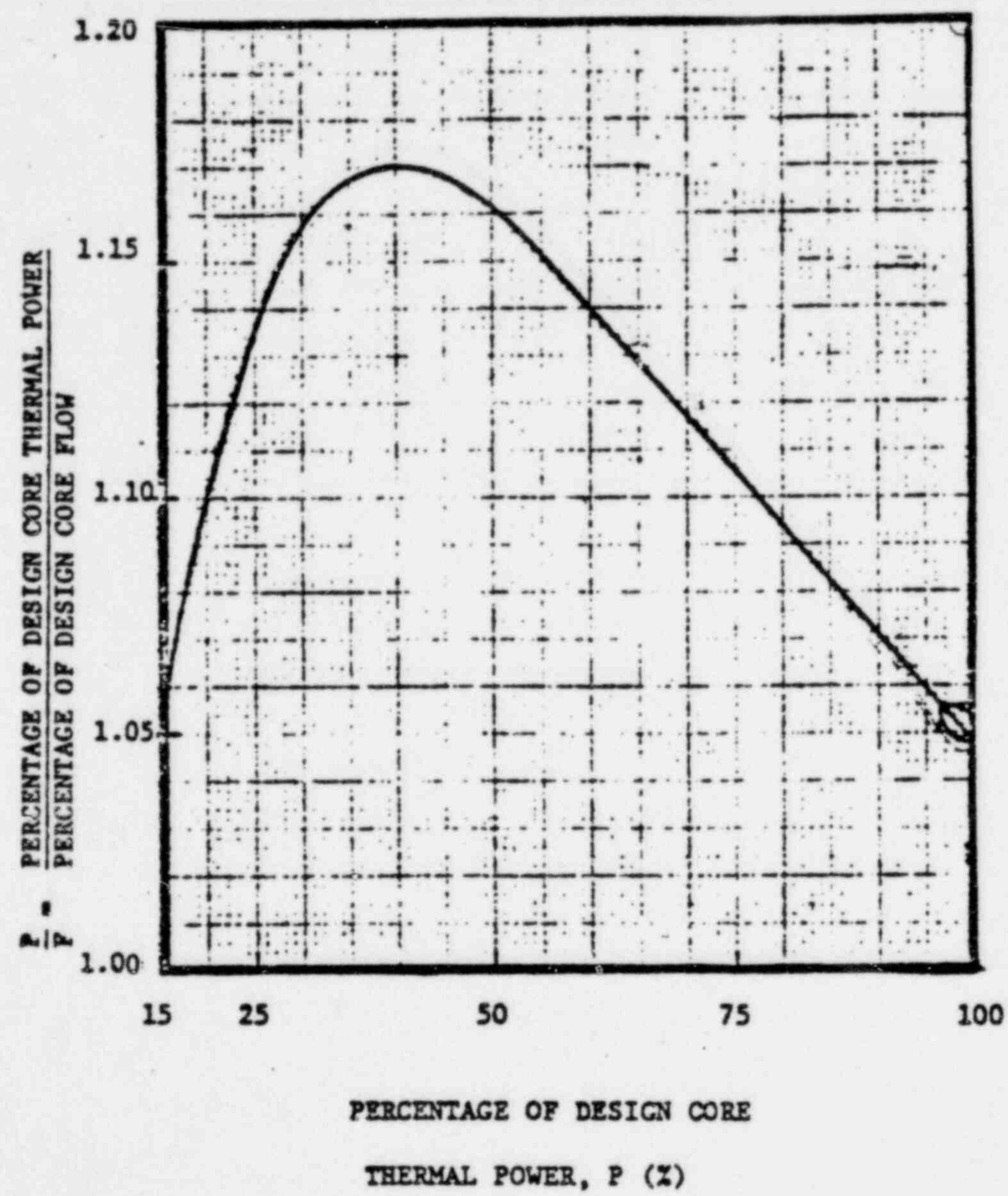

FIGURE 2.1

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.

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.

3.0 Staff Evaluation

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°
	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.

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.

3.1.2 Code Verification

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

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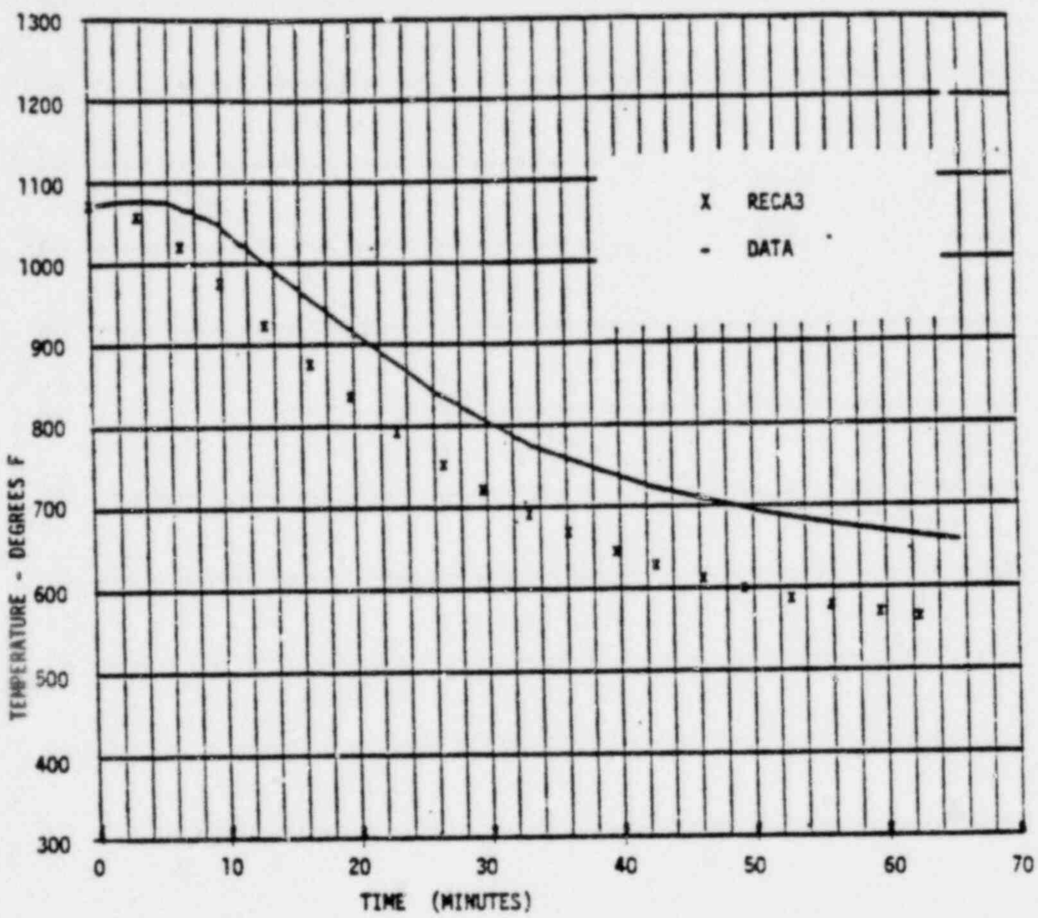
*The design bypass flow, or that flow which does not enter the core barrel, is 2.9 percent. The RECA analyses assume part of the unheated core flow as bypass. For these analyses, the bypass was input as 7.5 percent.



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

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.

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

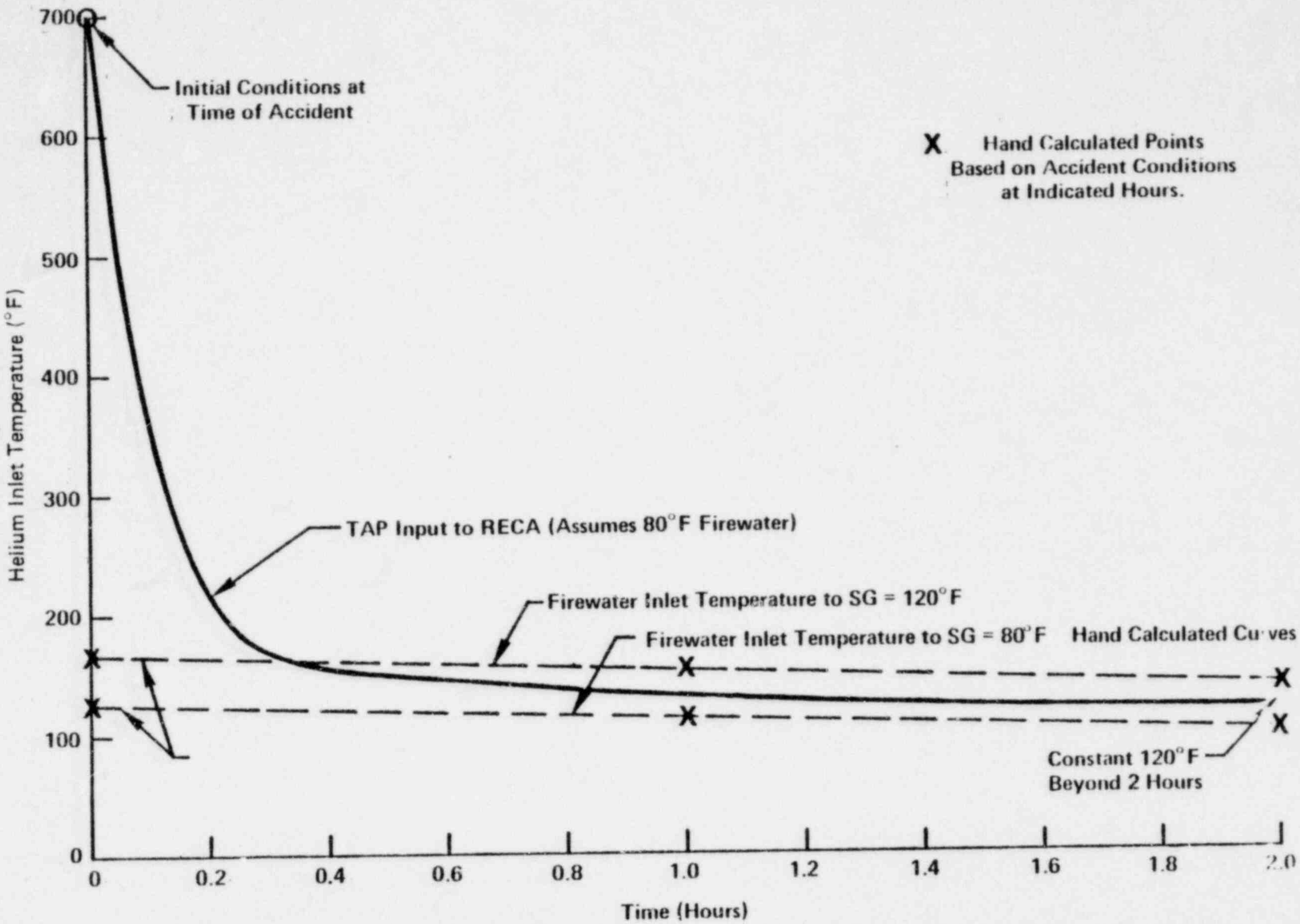


FIGURE 3.2

the exit to the steam generators will be approximately the same as the feedwater temperature, and therefore should not be a highly sensitve 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.

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.

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