XN-NF-86-146

# SUSQUEHANNA UNIT 2 CYCLE 2 SINGLE LOOP OPERATION ANALYSIS



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SUSQUEHANNA UNIT 2 CYCLE 2

SINGLE LOOP OPERATION ANALYSIS

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# 1.0 INTRODUCTION

This report presents the results of Exxon Nuclear Company's (ENC's) evaluation of system events in Single Loop Operation (SLO) for Susquehanna Unit 2 during Cycle 2 operation. In previous reload licensing reports ENC qualitatively concluded that two loop limits protect single loop operation except for the 0.01 increase in the MCPR Safety Limit to 1.07. This report confirms this conclusion quantitatively.

## 2.0 SUMMARY

Using ENC methodology and Cycle 2 core configuration the most limiting transient in SLO with regard to thermal margin is bounded by the 104% power and 100% flow generator load rejection without bypass valve operation (LRWB) transient. The Minimum Critical Power Ratio (MCPR) limits for potentially limiting single loop operation transients and for the bounding LRWB at full power and flow are shown in Table 2.1 for comparison. These transients were evaluated with all Susquehanna Unit 2 co-resident fuel types modeled and the most limiting type was used to determine the reported MCPRs.

The Safety Limit used to set the operating limit was confirmed as 1.07. This value was determined specifically for SLO conditions using ENC methodology.

A pump seizure accident from SLO was also evaluated and resulted in 243 rods calculated to attain boiling transition. If it is assumed that all rods which go through boiling transition fail, the results are less severe than those evaluated for other accidents in the Susquehanna FSAR. The LOCA/ECCS results are reported in XN-NF-86-125 which demonstrated that ENC MAPLHGRs developed for two loop operation remain applicable for single loop operation.

In conclusion, this report shows that two loop transient delta CPR's bound single loop transient delta CPR's. A comparison of the consequences of the pump seizure from single loop operation to previously reported FSAR evaluations of other events confirms that the radiological consequences of this event are well within 10 CFR 100 limits.

TABLE 2.1
THERMAL MARGIN

| Transient | Power/Flow | Δ CPR/MCPR |
|---|---|---|
| LRWB | 104/100 | 0.24/1.30 (1) |
| M-G Set Trip | 75.6/60.34 | 0.01/1.08 (2) |
| Pump Trip | 75.6/60.34 | 0.12/1.19 (2) |
| Idle Loop Start | 75.6/60.34 | 0.11/1.18 (2) |

Notes:
(1) Based on Safety Limit of 1.06
(2) Based on Safety Limit of 1.07
(3) All Transients analyzed using Technical Specification values for scram speed and scram delay.

# 3.0 TRANSIENT ANALYSIS

## 3.1 Analysis Bases

Single Loop Operation transients were analyzed using the plant transient simulation code COTRANSA (Ref. 1) and ENC methodology (Ref. 2). The delta CPRs were calculated using XCOBRA-T (Ref. 3). A comparison was made of COTRANSA results to the Susquehanna Unit 2 motor generator trip startup test to assure suitability of the code for single loop conditions.

MCPR limits established for full power and full flow are conservative for single loop operation because of the physical phenomena related to partial power and partial flow operation rather than modeling features. Under single loop conditions, full power conditions can not be achieved due to the physical capability of one operating recirculation pump. Thus, the MCPR limits established for two loop 100% flow are conservative for SLO.

The system conditions at which the SLO transients were evaluated are given in Table 3.1. In addition, the following conservative assumptions are made for all transients:

1. Technical specification scram speed.
2. Technical specification scram delay.
3. 110% multiplier on the integral power.

## 3.2 Load Rejection Without Bypass

The limiting system transient for the Susquehanna Units at rated conditions is the generator Load Rejection Without Bypass (LRWB) pressurization transient. In this transient the primary phenomena is the pressurization caused by abruptly stopping the steam flow by rapid closure of the turbine control valves. When the rapid pressurization reaches the core it causes a power excursion due to void collapse.

At reduced power and flow conditions there is a corresponding reduction in steam flow. With the lower steam flow the maximum pressurization of the core is reduced in comparison to rated conditions when the control valve is closed. The resulting power excursion and associated margin reduction are reduced below those of the full power case. Analysis has shown that the difference in the two loop operation and single loop operation core flow characteristics do not adversely effect the single loop operation case.

## 3.3 Feedwater Controller Failure

The second worst limiting transient at full power and flow is the Feedwater Controller Failure (FWCF) to maximum demand. This transient is also less severe at the power and flow conditions associated with single loop operation.

This transient assumes the feedwater controller fails to maximum demand and results in the maximum amount of subcooled feedwater in the downcomer. When this cooler water reaches the core the power rises. The core power rise is terminated by a reactor scram initiated by a turbine trip. The turbine trip is the result of the high water level trip caused by the additional amount of feedwater being injected.

At the reduced recirculation flows, the subcooling in the downcomer due to the high feedwater flow takes longer to transverse the core so that a high water level trip occurs before core power can rise as high as it does in the full flow case. As with the LRWB, the pressurization event resulting from the turbine trip is less severe for the reduced power in SLO.

## 3.4 Startup of the Idle Loop

During normal single loop operation the idle recirculation loop is within 50 degrees of the core inlet temperature prior to idle loop startup.

Normal procedure requires startup of the idle loop at a lower power than the maximum possible under SLO conditions and the idle fluid coupler at a setting of about 50% generator speed demand. The operating pump speed is then decreased to less than half of the rated flow, the idle pump is started and the flow adjusted to match adjacent loop flow. The idle loop discharge valve is opened and reactor power is adjusted as required.

The simulated idle loop startup transient conservatively assumed an abnormal sequence of operator action. The idle loop temperature was 100 degrees F lower than the core inlet temperature, reactor power and flow remained at their maximum SLO point, and the idle loop discharge valve was opened before the idle pump was started. The pump was started by starting the motor generator and allowing the starting pump to reach the same speed as the operating pump.

A graphical representation of important plant parameters is presented in Figure 3.1. The GE 8X8 fuel reaches its maximum delta CPR of 0.10 at 21.75 seconds and the ENC 9X9 fuel reaches its maximum delta CPR of 0.11 in 18.70 seconds. These values are bounded by the full power full flow MCPR limits.

### 3.5 Recirculation Pump Trip

Two cases of the recirculation pump trip were considered. The first was that of tripping the operating pump's motor generator set and allowing the combined motor generator and pump to coast down slowly. As is expected, this is a very benign transient with the maximum delta CPR for both fuel types being 0.01 at about 1.5 seconds into the transient.

The second case is that of tripping the recirculation pump itself and allowing it to coast down. The resulting coastdown is faster and therefore more limiting than the motor generator trip. The maximum delta CPR for the GE 8X8 fuel is 0.12 and occurs at 2.7 seconds. The maximum delta CPR for the ENC 9X9 fuel is 0.10 and occurs at 2.55 seconds. Figure 3.2 presents a graphical representation of significant plant parameters for this transient.

As is the case with the idle loop startup, the motor generator trip and the recirculation pump trip are bounded by the full power full flow LRWB.

TABLE 3.1
SLO REACTOR & PLANT CONDITIONS

| | |
|---|---|
| Reactor Thermal Power (75.67%) | 2490.83 $MW_t$ |
| Total Recirculating Flow (60.34%) | 60.34 Mlb/hr |
| Core Channel Flow | 55.01 Mlb/hr |
| Core Bypass Flow | 5.33 Mlb/hr |
| Core Inlet Enthalpy | 507.47 Btu/lb |
| Vessel Pressures | |
| Steam Dome | 994.40 psia |
| Upper Plenum | 998.17 psia |
| Core | 1004.80 psia |
| Lower Plenum | 1010.36 psia |
| Turbine Pressure | 965.41 psia |
| Feedwater/Steam Flow | 9.91 Mlb/hr |
| Feedwater Enthalpy | 335.16 Btu/lb |

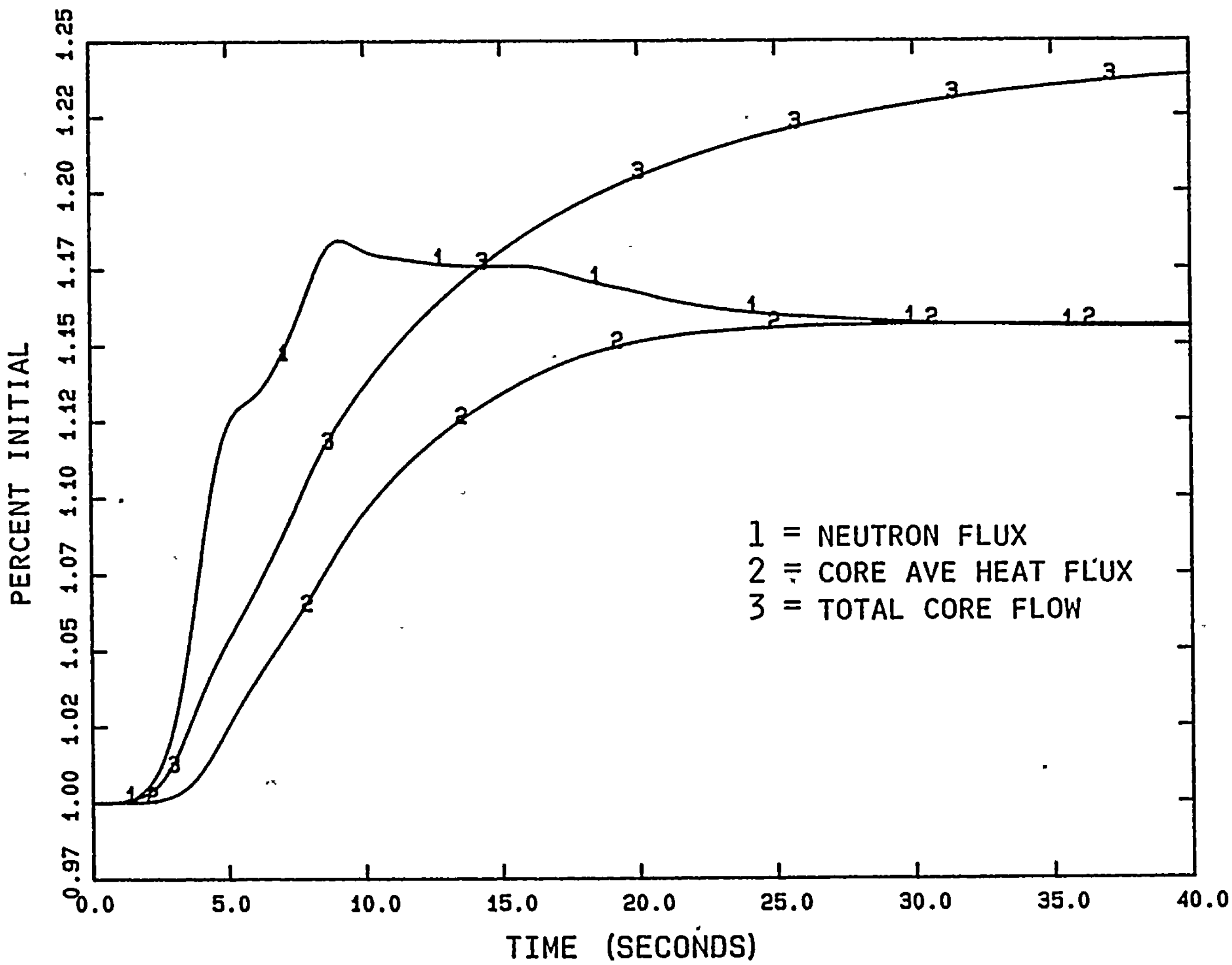


Figure 3.1 Startup of Idle Loop from SLO

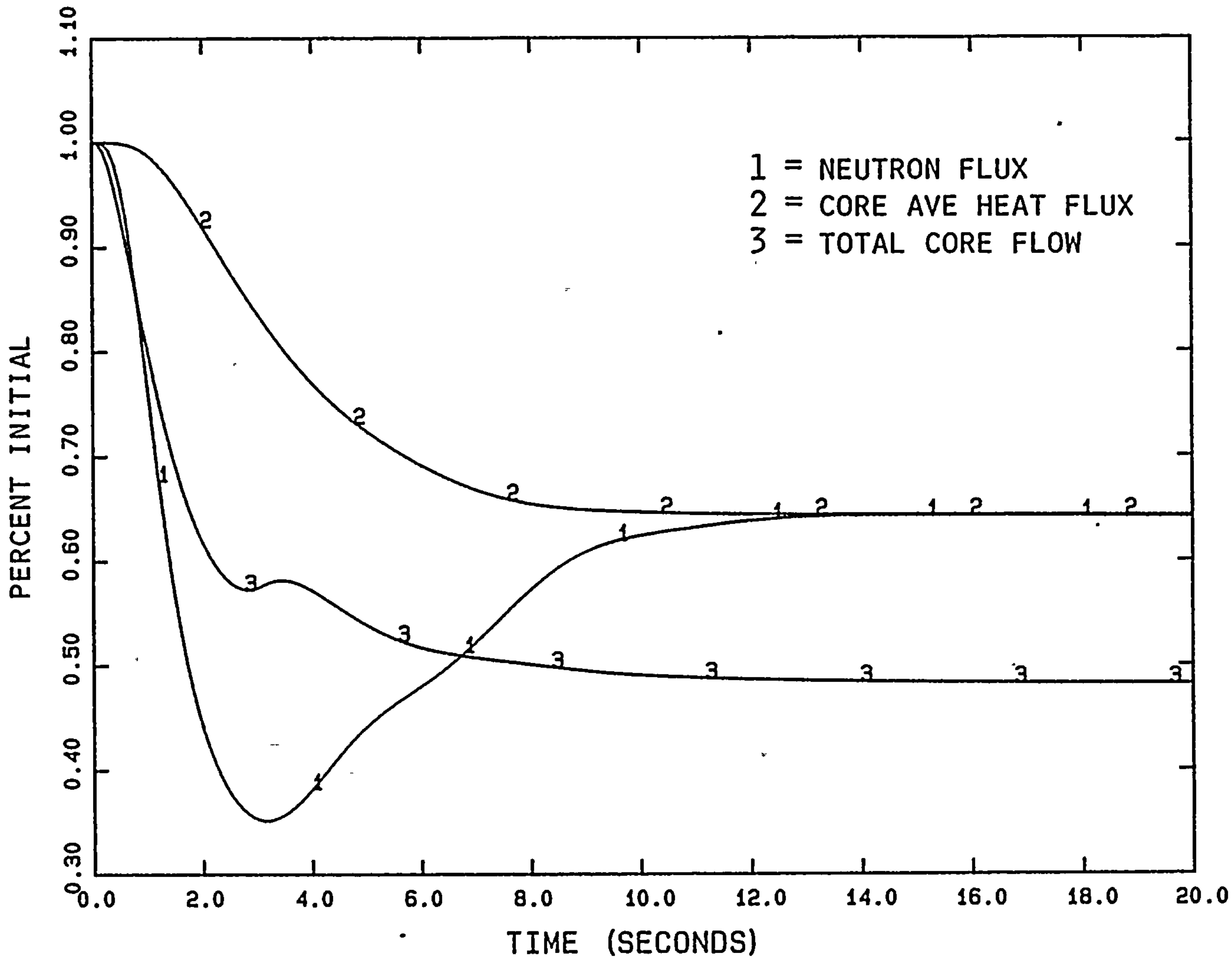


Figure 3.2 Pump Trip from Single Loop Operation

## 4.0 PUMP SEIZURE ACCIDENT

Pump seizure is a postulated accident where the operating recirculation pump suddenly stops rotating. This causes a rapid decrease in core flow, a decrease in the rate at which heat can be transferred from the fuel rods and a decrease in the critical power ratio. Analyses with COTRANSA and XCOBRA-T show the MCPR for ENC fuel would decrease by 0.33 during a pump seizure from single loop operation. ENC critical power methodology of Reference 4 showed that less than 243 fuel rods would fail during such an accident.

The COTRANSA code was used to simulate system response to a pump seizure in single loop operation from the conditions specified in Table 3.1. The operating recirculation pump rotor was stopped in 0.1 seconds causing a sudden decrease in active jet pump drive flow. At about 1.8 seconds the inactive jet pump diffuser flow went from negative flow to positive flow. In 8.0 seconds the dome pressure decreased to a minimum value of 973 psia and then started to increase again. Figure 4.1 presents a graphical representation of important system parameters during the transient.

The delta CPR for this event was calculated using XCOBRA-T. The ENC 9x9 fuel reached a maximum delta CPR of 0.33 at 2.20 seconds into the transient.

The methodology of Reference 4 was used to calculate the number of rods in boiling transition during the pump seizure accident. The following conservative assumptions were made:

- o The accident is initiated with the limiting fuel at the single loop MCPR operating limit of 1.31.

- The number of rods in boiling transition is equal to the number of rods in the entire core that would be in boiling transition at the steady state conditions corresponding to the conditions at 2.2 seconds when the minimum CPR is reached.

The system conditions at 2.2 seconds after initiation of a pump seizure accident were obtained from COTRANSA and XCOBRA-T. Voiding as a result of the decreased core flow caused the neutronic core power to decrease from the initial value of 2490 MWt to 655 MWt at 2.2 seconds. However, the thermal power being transferred from the surface of the hot channel corresponds to a core power of 2088 MWt at 2.2 seconds. A core power of 2088 MWt was used in the steady state analysis since this is power that is important to the coolant in the channel and boiling transition. Hydraulic demand curves were calculated to establish a relation from which assembly flow can be determined as a function of the fuel type and assembly power.

The power distributions used in this accident analysis were based on those used in the SLO safety limit analysis discussed in Section 5.0. For the conditions at the time when the minimum CPR was reached, 2.2 seconds, it was calculated that less than 243 rods will be in boiling transition. This number is based on a confidence level of 95%. Since boiling transition does not always result in rod failure, it can be conservatively stated that less than 243 rods will fail. A comparison of the consequences of the pump seizure from single loop operation to previously reported FSAR evaluations of other events confirms that the radiological consequences of this event are well within 10 CFR 100 limits.

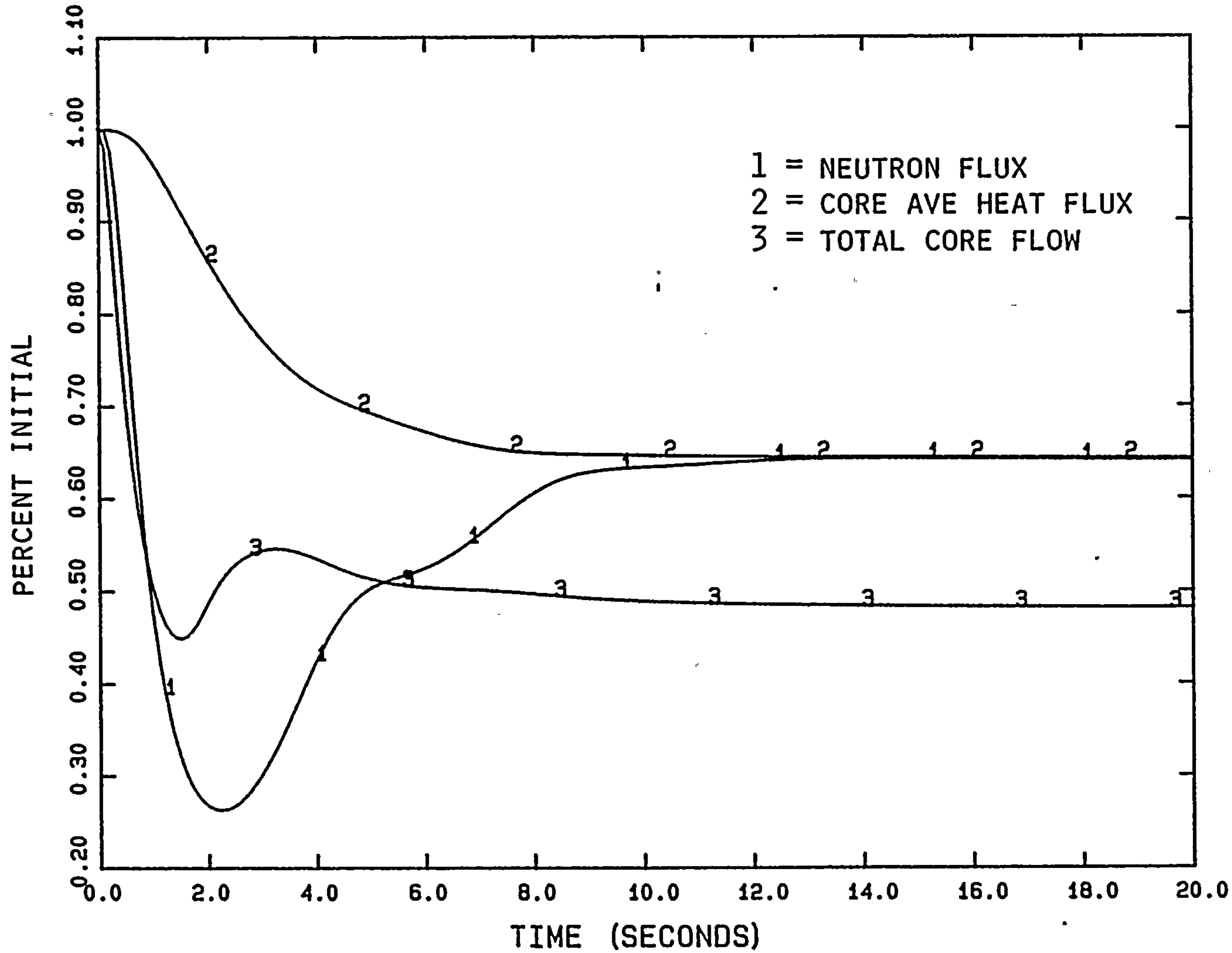


Figure 4.1 Susquehanna 2 Pump Seizure from Single Loop

## 5.0 SAFETY LIMIT

The MCPR fuel cladding integrity safety limit for single loop operation was calculated using the methodology described in Reference 4. In this methodology, a Monte Carlo procedure is used to evaluate plant measurement and power prediction uncertainties. The XN-3 correlation, Reference 5, is used to predict critical heat flux phenomena and non-parametric tolerance limits, Reference 6, are used to determine the expected number of rods in boiling transition.

During sustained operation at a MCPR of 1.07 with the design basis power distribution described below, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition at a confidence level of 95%. This supports a safety limit of 1.07.

The design basis power distributions used in this analysis for single loop operation were based on the predicted power distributions which were determined to be most severe during the two loop analysis, Reference 7. The radial power distribution was modified by increasing the maximum predicted radial power peaking factor to a value which corresponds to the single loop MCPR operating limit of 1.31 at the single loop operating conditions summarized in Table 3.1.

The uncertainties which were used in this single loop analysis were the same as the uncertainties which were used in the two loop analysis except for increases in the uncertainties for core flow, radial power, and axial power. These uncertainties were increased to account for increased uncertainties when the plant is in single loop operation, Reference 8.

## 6.0 REFERENCES


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