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1.0 INTRODUCTION

By letter dated February 22, 1979, as supplemented April 30 and May 1, 10, 18, 22, 23 and 25, 1979, Florida Power and Light Company (FPL or the licensee) requested an amendment to the St. Lucie Plant, Unit No. 1, Facility Operating License No. DPR-67 to accommodate Cycle 3 power operation. The principal changes from Cycle 2 operations and the Cycle 2 safety analysis are the following:

- The replacement of 68 spent fuel assemblies by fresh batch E and E* assemblies.
- An increase in cycle length from approximately 8600 MD/MTU in Cycle 2 to approximately 10,000 MWD/MTU in Cycle 3.
- Technical Specification (TS) changes required to accommodate the Cycle 3 core.
- Reanalysis of Design Basis Events (DBEs) to justify operation of the Cycle 3 core.

The following analytical methodologies were modified for the Cycle 3 core analysis:

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- Thermal-Hydraulic Credit for Statistical Combination previously credited to power distribution uncertainty factors is now applied to increase the allowed limit on Radial Peaking Factor (Fr).

- Peaking Factor Uncertainties of 6% in Fr and 7% in total core peaking factor (Fq) have been justified and adopted for Cycle 3.

- The coarse mesh neutronics code, ROCS, has been incorporated in the safety analysis.

- The method for computing Augmentation Factors has been modified to eliminate the over-prediction of power spiking inherent in the standard Combustion Engineering (CE) Augmentation Factor Model.

- The emergency core cooling system (ECCS) evaluation methodology has been modified to postulate two, rather than one, operable containment spray pumps.

In order to justify Cycle 3 operation the following special inspections were performed during the refueling outage:

- Inspection of CEA guide tube sleeves.

- Inspection of two CEAs which became stuck during Cycle 2 operation.

The CEA guide tube sleeving which was begun in Cycle 2 is being continued in Cycle 3.

2.0 BACKGROUND

St. Lucie Unit 1 ended Cycle 2 operation on April 1, 1979 with a total Cycle 2 burnup of 8385 MWD/MTU. St. Lucie Unit 1 operated through Cycle 2 with fuel batches A, B, C, and D at a licensed core power of 2560 MWD/MTU. Operation of Cycle 2 was at or near licensed power.

3.0 CYCLE 3 RELOAD

3.1 Core Design

The primary change to the core for Cycle 3 is the removal of the remaining 9 Batch A assemblies and 59 of the Batch B assemblies. These will be replaced by 40 Batch E assemblies and 28 Batch E* assemblies. The analysis for the Cycle 3 core was performed assuming a range of Cycle 2 burnups from 8000 to 8600 MWD/MTU to allow for flexibility in the Cycle 2 termination date. The actual Cycle 2 burnup of 8385 MWD/MTU is well within this range.

3.2 Fuel Design

3.2.1 Mechnical Design

The fresh batch E fuel is of essentially the same mechanical design as the batch D fuel introduced in Cycle 2 which we approved. Fuel assemblies of this design have had successful operating experience at Calvert Cliffs I, Fort Calhoun I, Millstone II and Maine Yankee.

The clad creep-collapse times for all fuel shown in the following table were computed using the CEPAN code (Reference T24).

Batch	Burnup for Clad Collapse (EFPH)*	Maximum Anticipated Burnup During Cycle (EFPH)*
B	≥ 24,600	23,690
C	≥ 24,600	23,690
D	≥ 24,200	14,170
E	≥ 26,900	7, 715

* 1 EFPH = 1.2830 MWD/MTU.

On the basis of the above information we find the mechanical design of the fuel acceptable.

3.2.2 Thermal Design

Using the FATES model (Reference T18), the thermal performance of the various fuel assemblies was evaluated, and the Batch E fuel was found to be limiting with respect to stored thermal energy. This is pertinent to the loss of coolant accident (LOCA) analysis of Section 9.1.

3.2.3 Chemical Design

The metallurgical requirements of the fuel cladding and the fuel assembly structural members for Batch E fuel have not been changed from the original Cycle 1 and Cycle 2 designs which have performed satisfactorily. The chemical or metallurgical performance of the Batch E fuel will be unchanged from that of the original core fuel and the discussions in the final safety analysis report (FSAR) are still valid. On this basis we find the chemical design of the Batch E fuel acceptable

3.2.4 Guide Tube Sleeving

Prior to the commencement of Cycle 2 those fuel assemblies which sustained substantial CEA guide tube wear in Cycle 1 had stainless steel sleeves installed in the guide tubes as a means of improving mechanical strength margins in worn areas. All Batch B, C. D and E fuel assemblies to be installed in CEA locations in Cycle 3 will also have stainless steel sleeves installed in the guide tubes in order to mitigate guide tube wear. A detailed discussion of the design of the sleeves and their effect on reactor operation is contained in Reference T27.

3.3 Nuclear Design

3.3.1 Fuel Management

The Batch E fuel is comprised of two sets of assemblies, each having a unique enrichment to minimize radial power peaking. The total Cycle 3 loading is 83.14 MTU. With this loading the Cycle 3 burnup for full power operation is expected to be between 9700 MWD/MTU and 10,100 MWD/MTU.

3.3.2 Nuclear Parameters

For the most part, the change from Cycle 2 nuclear parameters are small. The large increase in beginning of cycle, hot full power dissolved boron is due to greater U235 loading of the Cycle 3 core. This increased U235 loading is to accommodate the increased cycle length.

The values of physics parameters used in the safety analysis are, in all cases, either equal to or more conservative than the expected value or the limiting values in the TS.

3.4 Thermal Hydraulic Design

3.4.1 Thermal-Hydraulic Methodology

Steady state departure from nucleate boiling ratio (DNBR) analysis for Cycle 3 were performed using the same design codes as described in the FSAR. Appropriate adjustments were made to the inputs to these codes to reflect Cycle 3 parameters.

3.4.2 Thermal-Hydraulic Parameters

The Core Average Heat Flux, the Total Heat Transfer Area, and the Average Linear Heat Rate have changed slightly due to two effects: (1) The number of shims has decreased from 1296 in Cycle 2 to 588 in Cycle 3. The shims replace fuel pins, and hence there are a larger number of fuel pins in the Cycle 3 core. (2) Fuel densification is less severe in Cycle 3 than Cycle 2, leading to a greater fraction of the clad surface being available for heat transfer.

The pressure drop across the core and across the vessel have increased slightly because of the increased flow resistance of the control element assembly (CEA) guide tubes.

3.4.3 Thermal-Hydraulic Credit for Statistical Combination

In the DNB limit analysis, the assumed uncertainties in various measured parameters are not combined in a single equation but are factored into functional relationships as biases at various points in the analysis. This biasing of functional relationships throughout the analysis is equivalent to adding the absolute power uncertainties equivalent to the uncertainties in the various measured parameters and applying the total power uncertainty to the best estimate calculation. FPL has provided the following specific Cycle 3 uncertainties along with their equivalent power uncertainties:

Parameter	Parameter Uncertainty	Equivalent Power Uncertainty
ASI	0.06 ASIU	$\geq$ 2.2%
Pressure	22 PSI	$\geq$ 0.8%
Temperature	2 DEGF	$\geq$ 0.9%
Flow	4 %	$\geq$ 5.0%
Power (LSSS)	5 %	$\geq$ 3.5%
Power (LCO)	2 %	$\geq$ 1.4%

In the Cycle 3 analysis the equivalent sum of these uncertainties is 12.4 percent for the DNBR LSSS and 10.3 percent for the DNBR LCO. FPL believes that these components are statistically independent. We find this acceptable. This being the case, the proper method for combining these uncertainties is root sum square (RSS). The RSS combination yields 6.6 percent for the LSSS and 5.8 percent for the LCO, giving a net conservatism in the analysis of 5.8 percent for the LSSS and 4.5 percent for the LCO. For the Cycle 3 analysis FPL has proposed a partial credit of 3 percent for the LCO and LSSS. We find this acceptable.

In previous analyses of CE plants this same credit has been approved to offset the Fr measurement uncertainty. The FPL treatment differs from this in that the 3 percent credit has been applied instead to increase the allowed TS limit on Fr.

3.4.4 Thermal-Hydraulic (T-H) Effects of Guide Tube Sleeving

The T-H Effects of Guide Tube Sleeving were discussed by Northeast Nuclear Energy Company (NNEC) in Reference L1 and approved by us in Reference L3. The T-H effects for St. Lucie Unit 1 Cycle 3 are identical to those discussed by NNECO (Reference L2), and on the basis of our approval of the NNECO discussion, we find the T-H effects of Guide Tube Sleeving of St. Lucie Unit 1 Cycle 3 acceptable.

3.4.5 T-H Effect of Fuel Rod Bowing

The reduction in DNBR due to rod bowing is offset by a credit for low radial peaking in the critical assemblies. Fewer than 89 assemblies will exceed the NRC-determined penalty threshold burnup of 24,000 MWD/MTU (Reference T32) with a maximum burnup of 33,800 MWD/MTU. The corresponding DNBR penalty is 3.4%. The power distributions for Cycle 3 show the maximum radial peak for any of the 89 asssemblies to be at least 10% less than the maximum radial peak. Thus, the penalty is offset by the lower peaking of these assemblies and no power penalty for rod bowing is required for Cycle 3.

4.0 MODIFICATION OF ANALYTICAL METHODOLOGIES

4.1 PEAKING FACTOR UNCERTAINTY

In-core flux detector measurements are used to compute the core power distribution using the INCA code.

The INCA data reduction methodology and the technique for determining the required core and cycle specific coefficients are described in Reference T-19. Based on the review performed to date we find this portion of the CE INCA methodology acceptable for reloads.

The initial CE evaluation of peaking factor uncertainty was presented in CENPD-145 and CENPD-153 (References T19 and T20). In a meeting with CE on March 6, 1979, data was presented showing steady state measurement uncertainty of 6% in Fr and 7% in Fq to be conservative (Reference L10). In a previous meeting on October 12, 1978 CE had proposed that during load follow operations uncertainties of 9% in Fr and 10% in Fq should apply. Based on our review we find the above uncertainties acceptable for reloads.

For Cycle 3 operation the TS designate the appropriate peaking factor uncertainties as indicated above for steady state and load follow operations.

4.2 Use of ROCS COARSE MESH NEUTRONICS CODE

The Nuclear Design Model used in previous cycles has been based on PDQ (Reference T30), a neutron diffusion code which predicts power distribution

and fuel depletion with burnup. PDQ is a standard nuclear code accepted and used extensively throughout the industry. PDQ utilizies a simple low order differencing methodology, and because of this it is necessary to compute nuclear parameters at a large number of spatial points in order to achieve reasonable accuracy. In the Nuclear Design Model the core is represented by a two dimensional x-y cross section of the core with nuclear parameters being computed for each fuel pin, which corresponds to computing nuclear parameters of 196 points in each fuel assembly. This computational scheme is referred to as 2D PDQ. Three dimensional PDQ analyses are expensive, and hence are seldom performed.

Recently CE has begun performing some nuclear computations using the ROCS code (References T34 and T35) rather than PDQ. While CE has not yet submitted a ROCS topical report, FPL has supplied information on ROCS in the responses to requests for additional information on this reload (Reference L 12).

The ROCS code computes the same parameters as PDQ. By using a higher order differencing methodology than PDQ, ROCS is able to compute many parameters nearly as accurately as PDQ, having to compute nuclear parameters at only 1 or 4 points in the x-y plane for each assembly, rather than 196 as is done by PDQ. Also in PDQ the energy spectrum is divided into 4 energy ranges (called energy groups) whereas ROCS divides the energy spectrum into the equivalent

of one and one half energy groups (i.e., in the lower energy group only about half the nuclear parameters are represented). Because of the much coarser mesh in both space and energy allowed by ROCS, it is economically feasible to perform ROCS computations utilizing a three dimensional representation of the core, and in fact this is what is normally done.

For Cycle 3 the following safety parameters were computed using the ROCS code.

- Fuel Temperature Coefficients
- Moderator Temperature Coefficients
- Inverse Boron Worths
- Critical Boron Concentrations
- CEA drop distortion factors and reactivity worths
- Reactivity Scram Worths and Allowances
- Reactivity worth of regulating CEA bank
- Changes in 3-D core power distributions that result from inlet temperatures maldistributions (asymmetric steam generator transient)

None of these parameters requires the detailed knowledge of pin powers normally computed by 2D PDQ. CE states that in most cases these parameters are calculated more accurately by ROCS because of its ability to account for three dimensional effects. CE has stated that they observe guidelines to evaluate the adequacy of ROCS for computing these parameters on a case by case basis. If for any calculation the ROCS result is judged to be questionable, then the computation is done using PDQ.

Based on the above, we find the use of ROCS to be acceptable for this reload.

4.3 Modified Method for Computing Augmentation Factors

The densification of fuel pellets due to neutron exposure results in gaps in the fuel column. The gaps in a given fuel rod cause power spikes in nearby fuel rods and also power spikes above and below the gap in the fuel and containing the gap. CE has an NRC approved procedure for computing this power spiking which is described in Reference T18.

For this reload the licensee has identified an assumption in the CE methodology which causes the methodology to over-predict power spiking, and has proposed a revised model to give more realistic results. The assumption used by CE is that all fuel rods are equally susceptible to densification.

Test data shows, however, that the old low density fuel is more susceptible to densification than the fresh high density fuel. Furthermore, the high power density fuel rods, which are the only rods for which power spiking is a concern, are the fuel rods adjacent to the water holes within the fresh fuel assemblies. All these rods are in the immediate vicinity of only fresh fuel rods, the old fuel rods containing large gaps being at least two rod pitches away. The change the licensee has incorporated in his methodology is that he has postulated that all fuel rods less than two rod pitches away from the high power fuel

rods are fresh fuel rods with small gaps, and all fuel rods two or more rod pitches away are old fuel rods with large gaps.

We find this new methodology acceptable.

4.4 JUSTIFICATION FOR NOT RECOMPUTING TM/LP SETPOINT

The maximum Fr permitted by TS has increased by 3.9% from Cycle 2 to Cycle 3. Logically with an increase in Fr a recomputation of the thermal margin/low pressure (TM/LP) setpoint would be required. FPL has identified the following credits which more than offset the 3.9% increase in Fr, and hence assert that a recomputation of the TM/LP setpoint is not required.

- A partial credit of 3% was taken for the T-H statistical combination discussed in Section 3.4.3 of this SE
- There are fewer shims in the Cycle 3 core which increases the heat transfer area which, in turn, increases the DNBR margin by about 2%.
- There were some conservatisms in the Cycle 2 TM/LP setpoint computation.

We find this adequate justification for not recomputing the TM/LP setpoint.

5.0 SPECIFIC CYCLE 3 CONSIDERATIONS

5.1 STUCK CEA EVENT

Accordingly to Licensee Event Report 335-79-1 (Reference L7) on January 23, 1979, following a reactor trip performed as part of a shutdown procedure, two CEA stopped their fall into the core in the 8 inch withdrawn position. Subsequently they could not be moved by the CEA drive motors. It was necessary to remove the drive motors and manually pry the CEAs loose. After reconnecting the drive motors the CEAs were found to perform satisfactorily. The licensee states that there were no adverse consequences due to this occurrence.

The licensee states that the following examinations of these two CEAs were performed during the recent refueling outage:

(1) The 2 CEAs were withdrawn and reinserted twice with no indication of binding.

(2) The CEAs were thoroughly examined visually during the withdrawal/insertion operation. There was no indication of abnormal stress, wear, or binding.

(3) The withdrawal/insertion operation was performed with a load cell installed on each CEA. The load cell readings were normal.

(4) The top of each fuel assembly was visually examined. There was no indication of abnormal wear or binding.

(5) The top end of each guide tube sleeve was visually examined. Each sleeve was in place and there were no indications of deformation or damage.

(6) Both CEA assemblies were removed and transferred to the spent fuel storage pool for examination. Each CEA guide tube was vacuumed out their full length with vacuum filter examined for debris. No debris was found in any guide tube and no damage to fuel assemblies could be seen.

(7) The CEA's that stuck were dual CEA's and were both connected to the same drive mechanism in Cycle 2. Following the inspection in the fuel storage pool the assemblies were returned to different locations in the core and are now under CEA's which have separate single CEA drive units.

The licensee states that this examination gave no indication that the occurrence was related to guide tube wear, sleeving or loose parts. The licensee postulates that some debris may have temporarily collected in the lower dashpot section of a guide tube of one of the two CEA's affected. The guide tube necks down in the lower 12 inches and the sticking occurred at about 8 inches from the bottom. Therefore, it is possible that small amount of debris in the necked down section of a guide tube could cause CEA sticking. Since the reactivity in the last 8 inches of CEA travel is very small, the reactor could be safety shutdown even if all of the CEA's stuck at that level. Hot and cold drop time testing was performed, with all drop times well within limits and consistent with previous results. We find the licensee's inspection and corrective action to be adequate and concur that although the CEA sticking is not a result of an identifiable cause, it would not prevent safe shutdown of the reactor.

5.2 CEA GUIDE TUBE SLEEVE INSPECTION

Our evaluation of sleeving of CEA guide tubes at St Lucie Unit 1 is given in Section 7.0 of our safety evaluation of the cycle 2 reload, dated May 26, 1978 (Reference L4).

During the current refueling shutdown (cycle 3) the condition of guide tube sleeving was examined at St. Lucie Unit 1 by use of eddy current testing and TV scans. Similar eddy current tests and TV scans were accomplished at Millstone 2 and Calvert Cliffs 1 during refueling outages.

The eddy current tests of guide tube sleeving indicated that no detectable wear had occurred during the previous cycle at any one of the three units. Similarly, TV scans at the three units indicated proper sleeve seating. The eddy current tests at Calvert Cliff Unit 1, however, indicated inadequate crimp in some of the guide tube sleeves. Conversely, the eddy current tests at St. Lucie 1 indicated proper crimp in all guide tube sleeves. Pull tests on guide tube sleeves at Calvert Cliffs Unit 1 resulted in movement of sleeves in some of the guide tube sleeves which had shown inadequate crimp in eddy current tests. Pull tests were not performed at St. Lucie Unit 1 as there was no indication of an inadequacy of sleeve crimps during eddy current tests and no assemblies in Cycle 3 of the category where the problem has been detected (sleeved in the irradiated conditon).

The difference between the test results at St. Lucie Unit 1 and Millstone Unit 2 and those at Calvert Cliffs Unit 1 may be explained by the differences in installation of the sleeves.

At St. Lucie Unit 1 and Millstone Unit 2 pull tests were performed on the sleeves after the crimping step to verify the adequacy of the crimp. Following the "crimp verifiction" pull test, the sleeves were expanded into the guide tubes. At Calvert Cliffs Unit 1, however, the pull tests were not performed until after both the crimping and the expanding steps were completed. The licensee and CE have concluded that the method used at Calvert Cliffs Unit 1 added frictional resistance between the expanded sleeve and the guide tube to mask the presence of the inadequate crimp. The inadequate crimp would probably have been identified by an intermediate "crimp verification" pull test.

Therefore, based on the sleeving test evidence reported for St. Lucie Unit 1 and the method used for sleeving at St. Lucie we have determined that the sleeved CEA guide tubes are acceptable for cycle 3 operation. We have, however, requested that FPL provide an evaluation program which includes plans for inspections to determine guide tube wear experienced after the next cycle of operations with sleeved fuel assemblies.

6.0 REACTOR PROTECTION SYSTEM (RPS) ABNORMAL OPERATIONAL OCCURRENCES (AOOs)

The RPS AOOs, which are discussed in this section, are those postulated AOOs for which RPS trips other than the low RCS Flow Trip assure that the specified acceptable level design limits will not be violated. In the CE setpoint protection philosophy, it is postulated that complete protection rests on the combination of the limiting conditions for operation (LCO)s and limiting safety system settings (LSSS)s. However, for the transients discussed in this section, the LSSSs would provide a high degree of protection even if the LCOs were not observed.

The LCOs and LSSSs for the Cycle 3 TS were computed using the methods described in the CE SETPOINT METHODOLOGY REPORT (Reference T26). The transient plant responses used in the DBE reanalyses were calculated using the computer code CESEC (Reference T2). CESEC models both the primary and secondary coolant loops and accounts of the effects of all equipment responses in both loops.

The licensee stated that the need for reanalysis of a particular AOO is determined by comparison of the key parameters for that AOO to those of the last cycle for which a complete analysis was performed. If the key parameters are within the envelope of the reference cycle data, no reanalysis is required. A reanalysis might also be performed if it leads to a significant relaxation of TS restrictions. In some analyses, more conservative inputs than necessary are used in order that the analysis may be bounding for future cycles.

6.1 DNBR AND LHR CONSIDERATIONS

In the analysis of any postulated abnormal operational occurrence (AOO) it must be demonstrated that the DNBR limit would not be violated, i.e., that DNB does not occur, and that the linear heat rate (LHR) limit is not violated, i.e., that fuel centerline melt does not occur. In the analyses of accidents it is expected that the design limits would be violated, but the analysis must demonstrate that core damage is kept sufficiently small that the offsite exposure guidelines of 10 CFR 100 would not be violated.

CE has stated that most AOOs and Accidents are limited by DNBR considerations, and that generally the LHR values in the course of any event are so low that the LHR need not be considered in the analysis. The only events for which the LHR values become sufficiently high to merit consideration are the hot zero power (HZP) CEA Withdrawal, HZP Boron Dilution, CEA Drop, CEA Ejection and LOCA. We concur with CE's statement, and do not require LHR analysis for any except the five events cited.

6.2 TRANSIENT ANALYSIS INPUT TO SETPOINTS

6.2.1 TM/LP SETPOINT ADJUSTMENT

As stated in Reference T26, the computation of the TM/LP setpoint has as its starting point the computation of a TM/LP setpoint which guarantees that the DNBR does not decrease below 1.30 for steady state conditions. This steady state TM/LP setpoint cannot be used directly because during a transient the DNBR degrades for a short time after the trip setpoint is reached. To account for this, a time delay adjustment bias (TAB) is added to the steady state TM/LP setpoint so that the resultant dynamic TM/LP setpoint guarantees a trip soon enough in the course of the transient that core conditions will not degrade beyond the steady state TM/LP setpoint.

The TAB is determined by the transient for which the DNBR suffers the greatest degradation after the trip setpoint is reached. All potentially limiting transients must be considered in computing the TAB. Transients which do not require the TM/LP trip for production are excluded from consideration. With these criteria, the transients which must be considered in computing the TAB are the RPS AOOs which are initiated from full power, and steam generator tube rupture.

For a transient to be limiting in the determination of the TAB, it must produce a rapid degradation in DNBR. The rate of DNBR degradation was assessed, and it was determined the CEA Withdrawal Event and RCS Depressurization Event were more limiting than other competing events by a wide margin. The CEA

Withdrawal Event from Full Power produces the fastest temperature rise of any pertinent transient and the RCS depressurization event produces the fastest pressure decrease of any pertinent transient. Thus only these two events were considered in the computation of the TAB.

In the determination of the TAB from the analysis of those two events, the reactor trip is assumed to occur at the point of greatest rate of degradation of DNBR. For the purposes of computing the TAB this is a conservative approach. The minimum DNBR quoted in Table 6.0-1 for the CEA Withdrawal Event and RCS Depressurization Event is the minimum which would occur if the event were terminated by the TM/LP Trip. If some other trip is actuated before the TM/LP Trip, then the minimum DNBR would be higher. All transients other than these two are analyzed assuming the dynamic TM/LP setpoint determined by these two transients is in effect, and hence the minimum DNBR quoted for these transients in the reload application is appropriate.

6.3 RPS AOOs NOT REANALYZED

For the reasons explained in Section 4.4, even though Fr has increased in Cycle 3, there are credits which offset the resulting apparent DNBR degradation due to the Fr increase, and the demonstrated steady state DNBR margins have not degraded from their Reference Cycle values.

The Loss of Load Event produces a reactor trip in about 12 seconds in the FSAR analysis. In view of the fact that the CEA-DROP-TIME has increased by only 0.1 seconds, we have determined that the increase in CEA-DROP-TIME would have negligible impact on the Loss of Load Event.

We conclude that FPL has reanalyzed the correct set of RPS AOOS.

6.4 RPS AOOs REANALYZED

6.4.1 RCS Depressurization

The RCS Depressurization Event was reanalyzed because the CEA drop time has increased from its Reference Cycle Valve. As noted in Section 6.2, the purpose of performing this anlaysis is the computation of the TAB, which for the Cycle 3 RCS Depressurization Event is 30 psi. Since this is bounded by the 52 psi TAB computed for the hot full power (HFP) CEA Withdrawal event, the 52 psi TAB was used in computing the TM/LP setpoint. With the 52 psi TAB in effect, if the worst RCS Depressurization Event were terminated by the TM/LP Trip, the minimum DNBR would be approximately 1.34. We find this analysis and its predicted consequences acceptable.

6.4.2 HFP CEA Withdrawal

The CEA Withdrawal Events were reanalyzed because the CEA drop time has increased from its Reference Cycle value. As noted in Section 6.2 the purpose of performing the HFP analysis is the computation of the TAB, which for the Cycle 3 HFP CEA Withdrawal Event is 52 psi. Since this is greater than the 30 psi determined by the RCS Depressurization Event, the 52 psi TAB was used to determine the TM/LP Setpoint. We find this analysis and its resulting TAB acceptable. In Cycle 2 the limiting TAB was also 52 psi. In view of this fact and the discussion of Section 4.4 it is not necessary to adjust the TM/LP setpoint for Cycle 3 operation.

6.4.3 HZP CEA Withdrawal

The HZP CEA Withdrawal Event is terminated by the Variable High Power Trip at 25% power in the event of a rapid power escalation following criticality. The maximum power level attained is 148% of 2560 MWt, the maximum heat flux attained is 64.6% of 2560 MWt, and the minimum DNBR is 2.40.

The peak linear heat generation rate (PLHGR) in the course of the transient is approximately 24 KW/FT. CE states that for short transients (less than about 5 seconds) it is generally acceptable for the PLHGR to momentarily exceed the transient limit of 21 KW/FT so long as the average PLHGR for the transient is less than 21 KW/FT. For this event the power transient is approximately 3.5 seconds long and the average power is about 93% of full power. Thus the average PLHGR is well below even the steady state limit is 16 KW/FT, and the scoping criterion would indicate a low likelihood of fuel centerline melt.

The licensee has computed the total energy deposition in the fuel in the course of the transient and from this derived the peak fuel temperature. He has stated that fuel centerline melt is not predicted to occur.

We find this analysis and its predicted consequences to be acceptable.

7.1 AXIAL SHAPE INDEX (ASI) DNBR LCO ADJUSTMENT FOR TRANSIENT EFFECTS

The required overpower margin (ROPM) is a measure of the maximum DNBR degradation from the LCO which occurs in the course of a transient. The transients which must be considered in the computation of a limiting ROPM are those that are discussed in this section, all of which produce rapid degradation in DNBR.

In calculating the ROPM for these transients, the following input parameters are varied in cases where they have impact.

ASI
Power
Rod Insertion
Doppler Coefficient
Moderator Temperature Coefficient
Beta, Lambda, ℓ^*
RCS Pressure
Inlet Temperature
Axial Shapes Consistent With the Above Parameters

For each of a set of predetermined ASI values, the maximum ROPM for any of these transients and any combination of the other parameters is determined. An appropriate power to fuel design limit (PFDN) curve is chosen, which is a plot of ASI vs the power at which DNBR = 1.30. The limiting condition for operation for DNBR, based on the axial shape index, is determined by the appropriate combination of the power to fuel design limit and the required overpower margin.

There are certain transient effects which mitigate the DNBR degradation which are not factored into this analysis. Because of this the computer ROPM and this DNBR limiting condition for operation contain some conservatism above that normally identified.

We find this method of analysis and the resulting limiting condition acceptable.

The licensee states that for this reload the Four Pump Loss of Flow was the limiting transient for all values of the ASI. Thus the ROPM from the Four Pump Loss of Flow was used to compute the limiting condition for operation. This being the case, a Four Pump Loss of Flow Event initiated from any point will result in a minimum DNBR of 1.30, or slightly greater than 1.30 due to the conservatism cited above.

The Four Pump Loss of Flow analysis for this reload results in a minimum DNBR of 1.31, which is acceptable.

In the analysis of other AOOs considered in this section, the most conservative combination of initial conditions determined from the parametric study were used in the cases presented in the reload application. Hence the minimum DNBR given for these events is the minimum which can occur starting the event from any initial condition consistent with the limiting condition for operation.

7.2 AOOs USED TO ESTABLISH LCOs - NOT REANALYZED

The LCO sensitivity to the more important input parameters along with the actual changes in these parameters from the Reference Cycle values has been reviewed.

For the reasons explained in Section 4.4, even though Fr has increased in Cycle 3, there are credits which offset the resulting apparant DNBR degradation due to the Fr increase, and the demonstrated steady state DNBR margins have not degraded from their Reference Cycle values.

The Loss of AC Power Event described in the FSAR predicts a minimum DNBR of > 1.6. In view of the fact that the CEA-DROP-TIME has increased by only 0.1 seconds, we have determined that the increase in CEA-DROP-TIME would have negligible impact on the Loss of AC Power Event.

Based on Table 7.2-1 and the above arguements, we conclude that FPL has reanalyzed the correct set of AOOs.

7.3.1 Four Pump Loss of Flow

The Four Pump Loss of Flow Event was reanalyzed because the CEA Drop Time, the Radial Power Peaking, and the Axial Power Peaking have increased from their Reference Cycle values. This analysis was used to determine the LCO as described in Section 7.1. This event results in a minimum DNBR of 1.31. We find this analysis and its resulting LCO acceptable.

8.0 POSTULATED ACCIDENTS OTHER THAN LOCA

In postulated accidents a certain amount of plant and/or core damage is expected. As noted in Section 6.1 the purpose of the accident analysis is to demonstrate that in the course of an accident the offsite exposure guidelines of 10 CFR 100 are not exceeded.

8.1 ACCIDENTS NOT REANALYZED

We have reviewed the safety parameter inputs to the FPL's accident analysis and conclude that FPL has selected the correct set of accidents to reanalyze.

8.2 ACCIDENTS REANALYZED

8.2.1 CEA EJECTION

The CEA Ejection Events were reanalyzed because the CEA Drop Time and the Azimuthal Power Tilt Allowance have increased from the Reference Cycle values. Also the Cycle 3 pin census is more adverse.

The CEA Ejection Event is very rapid, and the total energy deposition is the proper parameter to monitor, rather than LHR or DNBR as is the case for other transients. The specific criteria employed are:

- Clad Damage Threshold
  Total Average Enthalpy = 200 calories/gram.
- Incipient Centerline Melting Threshold
  Total Centerline Enthalpy = 250 calories/gram.
- Fully Molten Centerline Threshold
  Total Centerline Enthalpy = 310 calories/gram.

Both the HFP and HZP cases were analyzed, and it was found that the HFP case was more limiting. For this case the analysis performed by the licensee predicted a maximum total centerline enthalpy of 289.0 calories/gram with

2.8% of the pins suffering incipent centerline melting. In the Reference Cycle the maximum total centerline enthalpy was 293.0 calories/gram, but because of the less adverse pin census in this case the fraction of fuel suffering incipent centerline melting was only 1.3%. The predicted total radial average enthalpy is less than 200 calories/gram, and hence the licensee's analysis assumes no clad damage and no radioactive release from fuel pins suffering incipient centerline melting. We find this analysis and its predicted consequences to be acceptable.

8.2.2 Siezed Rotor

The Siezed Rotor Event was reanalyzed because the radial peaking factors have increased and the pin census has become more adverse than the Reference Cycle values. In the Siezed Rotor Event, it is normal to predict a small fraction of the fuel will suffer DNB, and hence fail, and the purpose of the analysis is to assure that the radioactive release which would result from this accident is acceptable. The Cycle 3 analysis of the Siezed Rotor event predicts 0.99% cladding failure through DNB.

While we have no analysis for the radioactive release for the Siezed Rotor Event, the release can be compared to release from the CEA Ejection Event for which we do have an analysis which is acceptable. The CEA Ejection Event was previously analyzed assuming 10% failed cladding and assuming that

all radioactive release was via the path from primary coolant to secondary coolant to atmosphere, which is the same as the release path for the Siezed Rotor Event. Thus the radioactive releases for the Siezed Rotor Event with 0.99% failed fuel should be about one tenth that calculated for the CEA Ejection Event. Therefore, radioactive releases for the Siezed Rotor Event are within the 10 CFR 100 guidelines by a large margin. We find this analysis and its predicted consequences acceptable.

9.0 LOCA ANALYSIS

9.1 LARGE BREAK LOCA

The cycle 2 results for the Large Break LOCA were as follows (Reference 1):

	PCT (DEGF)	Cladding Oxidation (%)	Clad Rupture Time
Accepted Limit	2200	17	
Low Density Fuel	1952	7.63	Blowdown
High Density Fuel	1975	10.4	Reflood

The more pertinent LOCA input parameters for Cycle 2 and Cycle 3 are as follows:

Parameter	Units	Value for Limiting Fuel	
		Cycle 2 Batch D	Cycle 3 Batch E
Average LHR	KW/FT	6.2126	6.0956
Gap Conductance of PLHGR	$\frac{BTU}{HR\text{-}FT^2\text{-}DEGF}$	1552	1525
Fuel Centerline Temp at PLHGR	DEGF	3484	3512
Fuel Average Temp at PLHGR	DEGF	2184	2197
Hot Rod Gas Pressure	PSIA	1047.8	1031
Hot Rod Burnup	MWD/MTU	820	820

Although the stored energy for the limiting fuel for Cycle 3 is slightly higher than the limiting fuel for Cycle 2, the licensee states that the ECCS performance for Cycle 3 remains bounded by the previous cycle for the following reasons:

1. The cycle 2 analysis did not utilize the PARCH code (Reference T16) to compute steam cooling heat transfer coefficients after the rod rupture is predicted to occur. Since the peak clad temperature is achieved during late reflood, use of the PARCH code would have significantly reduced the peak temperatures calculated in the Cycle 2 analysis. The licensee states that use of the PARCH methodology should also reduce the calculated peak clad temperature by at least 100 DEGF for Cycle 3.

2. The radial power distribution for use in the thermal rod-to-rod radiation model is less limiting for cycle 3 than that used for the previous cycle. Use of the less limiting radiation enclosure for cycle 3 should reduce the peak clad temperatures at least 50 DEGF.

The above defined margins are far in excess of the 13 DEGF increase in fuel stored energy for the Cycle 3 limiting fuel relative to the previous cycle. The licensee concludes, therefore, that the cycle 2 analysis is also applicable to Cycle 3. In addition, we have revised the proposed TS to reduce the allowable peak LHR (Figure 3.2-1 in TS) from 14.8 KW/FT to 14.68 KW/FT. Also, we have revised Figure 3.2-2 (Axial Shape Index) to reduce the fraction of maximum allowable power level from 0.84 to 0.825. FPL has agreed to these revisions. Our calculations show that this will reduce the fuel average temperature at peak LHR from 2197 F to 2184 F which would be the same as Cycle 2. Therefore, this assures that the Cycle 3 operation is bounded by Cycle 2 analysis and is in full conformance with 10 CFR Part 50 Appendix K.

9.2 SMALL BREAK LOCA ANALYSES

On January 4, 1974 the Atomic Energy Commission issued new Acceptance Criteria for the ECCS for Light-Water Cooled Reactors (Reference T33). Subsequently CENPD-137 (Reference T15) was submitted to the AEC which demonstrated that for Calvert Cliffs class plants (2560 MWt), breaks smaller than 0.5 FT2 are not limiting. We still find this analysis acceptable, and hence the Small Break LOCA requires no reanalysis for Cycle 3.

9.3 CE RUPTURE STRAIN MODEL

The Rupture-Strain curve is a plot of Clad Strain (Clad Swelling) vs Clad Temperature at the point of clad rupture in a LOCA Event. The Rupture-Strain curve is an integral part of the CE ECCS flow blockage model. Recently the NRC staff has determined that, for clad rupture which occur during the reflood phase of the LOCA, the Rupture-Strain curve used by CE is nonconservative. That is, the CE cladding rupture-strain curve used in the flow blockage model underestimates the degree of cladding swelling in a certain portion of the curve. As a result, we informed CE of this and asked them to recompute the effect of using a rupture strain model that was conservative relative to specified portions of the relevant data and that would be acceptable under Appendix K. CE then performed some reanalysis with this curve that is conservative for System 80 plants, and they also included an improved flow distribution/heat transfer model in the vicinity of the blockage. CE has performed a calculation with these two modifications for a worst case break for System 80 plants.

We are still reviewing the new flow blockage model. We plan to complete this review by June 1979. However, we have sufficiently reviewed the information to conclude that the rupture-strain model proposed by CE is acceptable for applications to System 80 plants. Based on our review to date the flow distribution/heat transfer model used in the calculation for the System 80 worst case break is acceptable with respect to Appendix K.

System 80 plants predict clad rupture during the reflood phase of the LOCA as does St. Lucie 1. Accordingly, it is our judgment that calculations for St. Lucie 1 made with either the current model or using the revised rupture-strain curve and revised crossflow and heat transfer model, would yield comparable results. Therefore, we conclude that the St. Lucie 1 ECCS performance has been shown to conform to the applicable requirements of Appendix K to 10 CFR 50.

9.4 CONTAINMENT SPRAY ECCS ERROR

The previous ECCS analysis (Reference L1) postulated only one Containment Spray Pump operable during the reflood phase of a LOCA. Further investigation revealed that a more conservative and realistic result would be obtained if it were postulated that both Containment Spray Pumps are operable.

During the reflood the whole RCS is at containment pressure plus or minus whatever pressure is exerted by the heights of the columns of water in the RCS. The sole driving force reflooding the core is the height of water in the vessel downcomer, and the principal resistance to reflood is steam binding in the SGs. If the usual cold leg break is assumed the only path by which steam can exit the core is via the SGs. The mass flow through the SGs is proportional to the square root of the steam density, and being at essentially the same pressure as the containment, the steam density is inversely proportional to the containment pressure. Thus the lower contaiment pressure which results from having two containment spray pumps operable decreases the steam mass flow and hence decreases the reflood rate.

The previous ECCS analysis was performed based on heat transfer coefficients corresponding to the very conservative PLHGR of 17 KW/FT. The new analysis, which is performed assuming both Containment Spray Pumps are operable, assumed the TS PLHGR of 14.8 KW/FT. The new analysis is only a small segment of the total analysis intended to demonstrate the sensitivity to changing these two parameters. This new analysis predicted a PCT of 2022 DEGF, as compared with 2035 DEGF for the old analysis. Thus the combination of the two more realistic assumptions predicts a drop of 13 DEGF in PCT. We find this method for demonstrating that the original analysis is bounding to be acceptable.

10.0 TS CHANGES

The proposed Cycle 3 TS changes and the rationale for these changes are presented below.

Page 1-6 Section 1.30	The definition of LOAD FOLLOW OPERATION is introduced. This is in keeping with a change approved by us in other recent CE reload applications. The purpose of this definition is to provide a basis for determining when the higher uncertainty of TS 4.2.1.4.b.2 shall apply.
Page 2-4 Table 2.2-1	The Containment Pressure-High Trip Setpoint is changed from 3.9 psig to 3.3 psig. This is to correct a typographical error. As indicated in Reference L6, the analysis performed predicted a trip value of 3.3 psig and 3.3 psig appears in the FSAR, but 3.9 psig was inadvertently typed in the TS for Cycle 2.
Page 2-4 Table 2.2-1 Page B2-5 Section 2.2	The units of the Steam Generator Pressure-Low Trip Setpoint are changed from psig to psia. Thus, the value is changed from 485 psig to 500 psia.

Page 2-7 Figure 2.2-2 Page 3/4 2-4 Figure 3.3-2 Page 3/4 2-8a Figure 3.2-3a Page 3/4 2-8b Figure 3.2-3b Page 3/4 2-15 Figure 3.2-4	These figures are the LHR ASI LSSS, the LHR LCO, and the DNBR LCO. These were recomputed using Cycle 3 parameters. The LHR ASI LSSS was computed using the methods described in the CE SETPOINT METHODOLOGY Topical Report, Reference T26. In the treatment described in the CE SETPOINT METHODOLOGY REPORT, the two LCOs are maximum allowed power values given as a function of ASI. The current CE methodology is a generalization of this concept in which the maximum allowed power is given as a function of both ASI and Radial Power Peaking. We understand this concept and find the current methodology acceptable for this reload. For Cycle 3, the tents of Figures 2.2-2, 3.2-2 and 3.2-4 are more limiting than the corresponding Cycle 2 tents to compensate for the increased allowed Radial Power Peaking and the increased Azimuthal Power Tilt Allowance.
Page 3/4 1-26 Section 3.1.3.4 Page B3/4 1-4 Section 3/4.1.3	The allowed CEA drop time to 90% insertion has been increased from 3.0 seconds to 3.1 seconds. This was done in order to increase conservatism and to bound future cycle increases in CEA drop time. FPL has stated that in the most recent surveillance the longest measured CEA drop time was less than 3.0 seconds.

Page 3/4 2-1 Section 4.2.1.3	For monitoring LHR via the Excore Detectors, a requirement has been added that the CEAs shall be withdrawn at least to the Long Term Steady State Insertion Limit. This requirement is added because the LHR ASI LCO is derived assuming that the CEAs are not inserted beyond the Long Term Insertion Limit, and the LCO is not valid for the case where the CEAs are inserted farther.
Page 3/4 2-2 Section 4.2.1.3.c.2	The reference to Figure 3.2-3 is changed to Figure 3.2-3a to reflect the renumbering of TS figures.
Page 3/4 2-6 Section 3.2.2	
Page 3/4 2-2 Section 4.2.1.4.b.2	The Measured Power Distribution Uncertainty is changed to reflect the discussion of Section 4.1 of this SE.
Page B3/4 2-1 Section 3/4 2-1	
Page 3/4 2-3 Figure 3.2-1	This figure is changed to remove cycle specific details in order to extend its applicability beyond Cycle 2.
Page 3/4 2-5 Figure 4.2-1	This is a figure of Augmentation Factors vs Distance from Bottom of Core. It was computed using specific Cycle 3 parameters using the methods described in Reference T18 modified as indicated in Section 4.3 of this SE.

Page 3/4 2-6 Section 3.2.2	F_{xy}^T is increased from 1.589 to 1.627. This is consistent with the revised Figure 3.2-3a. Since this increase is properly incorporated in the Safety Analysis, we find this change acceptable.
Page 3/4 2-6 Section 4.2.2.2	In the load follow mode the measured value of F_{xy}^T is increased by 3% to account for the additional uncertainty indicated in Section 4.1 of this SE.
Page 3/4 2-6 Section 4.2.2.2 Page 3/4 2-9 Section 4.2.3.2 Page 3/4 2-11 Section 3.2.4	The Azimuthal Power Tilt Allowance has been increased from 2% to 3%. Since this increase is properly incorporated in the Safety Analysis we find this change acceptable.
Page 3/4 2-9 Section 3.2.3	The measured value of F_r^T has been increased from 1.563 to 1.64. This is consistent with the revised Figure 3.2-3b. Since this increase is properly incorporated in the Safety Analysis we find this change acceptable. Also in this section the reference to Figure 3.2-3 is changed to Figure 3.2-3b to reflect the renumbering of figures.

Page 3/4 2-9 Section 4.2.3.2	For the load follow mode the measured value of F_r^T is increased by 2% to account for the additional uncertainty indicated in Section 4.1 of this SE.

We have reviewed all TS changes and find them acceptable.

11.0 STARTUP PHYSICS TESTING

We have reviewed the Physics Startup Test Program as described in Reference L14. This program includes:

1. Rod Drop Times.
2. CEA Criticality Measurements.
3. CEA Symmetry Checks.
4. Unrodded Critical Boron Concentration Measurements.
5. Moderator Temperature Coefficient Measurements.
6. Rod Worth Measurements.
7. Shutdown Margin Verification.
8. Power Distribution Measurements at 30%, 50% and 100% Power.

Additional information as to Review and Acceptance Criteria and required Remedial Action was requested. This information was submitted in Reference L19. We have reviewed the entire Physics Startup Test Program and find it to be acceptable.

12. LICENSE CONDITION D - NEUTRON SHIELDING

By letter dated November 9, 1978, FPL notified the NRC that the additional neutron sheidling for the reactor cavity had been installed as required by License Condition D of Enclosure 1 to License No. DPR-67. The design of the shielding modification was previously reviewed and found acceptable by the NRC (Reference L4). Because the shielding is now installed, the License Condition D of Enclosure 1 may be deleted.

13.0 ENVIRONMENTAL CONSIDERATION AND CONCLUSION

13.1 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.

13.2 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probabilty or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, 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.

Dated: May 27, 1979

14.0 REFERENCES AND METHODOLOGY CROSS INDEX

14.1 LETTER REFERENCES

L1 NNECO submits MNPS-2 Cycle 2 Reload Application, Donald C. Switzer (NNECO) to George E. Lear (NRC), September 2, 1977.

L2 FPL submits Cycle 2 Reload Application, Robert E. Uhrig (FPL) to Victor Stello (NRC), March 22, 1978.

L3 NRC approves MNPS-2 Cycle 2 Reload Application, Robert W. Reid (NRC) to Donald C. Switzer, April 19, 1978.

L4 NRC approves Cycle 2 Reload Application, Robert W. Reid (NRC) to Robert E. Uhrig (FPL), May 26, 1978.

L5 CE presents analysis to defend their Rupture-Strain ECCS Model, A.E.Scherer (CE) to Denwood F. Ross (NRC), September 18, 1978.

L6 FPL submits revision to ECCS performance results of Reference L1, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), November 9, 1978.

L7 FPL proposes TS changes, Robert E. Uhrig (FPL) to Victor Stello (NRC), January 8, 1979.

L8 FPL reports stuck CEA occurrence, A.D.Schmidt (FPL) to James P. O'Reilley (NRC), February 22, 1979.

L9 FPL submits Cycle 3 Reload Application, Robert E. Uhrig (FPL) to Victor Stello (NRC), February 22, 1979.

L10 CE provides slides on Power Measurement Uncertainties from meeting of March 6, 1979, A.E.Scherer (CE) to Paul S. Check (NRC), March 7, 1979.

L11 NNECO provides additional information on MNPS-2 Cycle 3 Reload Application, W. G. Counsil (NNECO) to Robert W. Reid (NRC), March 27, 1979.

L12 FPL provides additional information on Cycle 3 Reload Application, Robert E Uhrig (FPL) to Robert W. Reid (NRC), April 30, 1979.

L13 BGE provides additional information on CC-1 Cycle 4 Reload Application, A. E. Lundvall (BGE) to Robert W. Reid (NRC), May 7, 1979.

L14 FPL provides information on Cycle 3 Startup Testing, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 10, 1979.

L15 FPL provides additional information on Cycle 3 Reload Application, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 18, 1979.

L16 FPL provides additional information on Cycle 3 Reload Application, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 22, 1979.

L17 FPL provides additional information on Cycle 3 Reload Application, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 23, 1979.

L18 FPL provides information on CEA Guide Tube Sleeving Inspection, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 23, 1979.

L19 FPL provides information on Cycle 3 Startup Testing, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 25, 1979.

14.2 TOPICAL REFERENCES AND REVIEW STATUS

REF. No.	REFERENCE TITLE and REVIEW STATUS
T1	CENPD-98, COAST Code Description," April 1973. Review Status: Approved.
T2	CENPD-107, "CESEC - Digital Simulation of a C-E Nuclear Steam Supply System," April 1974. Review Status: Still under review. Initial NRC comparison of CESEC computations with RELAP-3 computations show good agreement. Considered to be acceptable for reload analyses.
T3	CENPD-132, "Calculative Methods for the CE Large Break LOCA Evaluation Model," August 1974. Review Status: Approved.
T4	CENPD-132, Supplement 1, "Updated Calculative Methods for the CE Large Break LOCA Evaluation Model," December 1974. Review Status: Approved
T5	CENPD-132, Supplement 2, "Calculational Methods for the CE Large Break LOCA Evaluation Model," July 1975. Review Status: Approved.

T6 CENPD-133, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis," April 1974.

Review Status: Approved.

T7 CENPD-133, Supplement 1, "CEFLASH-4AS, A Computer Program for Reactor Blowdown Analysis of the Small Break Loss-of-coolant Accident," August 1974.

Review Status: Approved.

T8 CENPD-133, Supplement 2, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis (Modification)," December 1974.

Review Status: Approved.

T9 CENPD-134, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," April 1974.

Review Status: Approved.

T10 CENPD-134, Supplement 1, "COMPREC-II, A Program for Emergency Refill-Reflood of the Core (Modification)," December 1974.

Reveiw Status: Approved.

T11 CENPD-135-P, "STRIKIN II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.

Review Status: Approved.

T12 CENPD-135, Supplement 2-P, "STRINKIN- , A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modification)," February 1975.

Review Status: Approved.

T13 CENPD-135, Supplement 4-P, "STRINKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.

Review Status: Approved.

T14 CENPD-135, Supplement 5-P, "STRINKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.

Review Status: Approved.

T15 CENPD-137, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," August 1974.

Review Status: Approved

T16 CENPD-138, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.

Review Status: Approved.

T17 CENPD-138, Supplement 1, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup" (Modification), February 1975.

Review Status: Approved.

T18 CENPD-139, "CE Fuel Evaluation Model," July 1974.
Review Status: Approved.

T19 CENPD-145, "INCA: Method of Analyzing In-Core Detector Data in Power Reactors," Combustion Engineering, April 1975.
Review Status: Still under review. INCA computational methodology appears to be reasonable the uncertainty analysis in these topical's is not acceptable. Based on subsequent discussion and submittals, the interim assumed measurement uncertainties of 6% in Fr and 7% in Fq appear to be reasonable.

T20 CENPD-153,"Evaluation of Uncertainty in the Nuclear Form Factor Measured by Self-Powered Fixed In-Core Detector Systems", Combustion Engineering, August 1974.
Review Status: Same as that for T19.

T21 CENPD-161-P, "TORC Code - A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975.
Review Status: Approved.

T22 CENPD-162-P-A, "CE Critical Heat Flux," September 1976.
Review Status: Approved.

T23 CENPD-183, "CE Methods for Loss of Flow Analysis," July 1975.
Review Status: Review has progressed to the point where this methodology is considered to be acceptable for reload analyses.

T24 CENPD-187, "CEPAN Method of Analyzing Creep Collapse of Oval Cladding," June 1975.

Review Status: Approved.

T25 CENPD-190A, "CE Ejection, C-E Method for Control Element Assembly Ejection," July 1976.

Review Status: Approved.

T26 CENPD-199-P, "C-E Setpoint Methodology," April 1976.

Review Status: Review has progressed to the point where this methodology is considered to be acceptable for reload analyses.

T27 CEN-80(N)-P, "Millstone Unit 2 Reactor Operation With Modified CEA Guide Tubes," February 8, 1978.

T28 CEN-89(F)-P, "Solution to Increased Water Hole Peaking in Operating Reactors St. Lucie 1, P. C. Uhrig to R. Reid, April 10, 1978.

Review Status: New TM-LP methodology appears reasonable. Uncertainty analysis presented is not acceptable, but has been superceded by subsequent analysis by CE, as indicated for T19 and T20 above.

T29 WAPD-TM-479, J. A. Redfield, "CHIC-KIN - A Fortran Program for Intermediate and Fast Transients in a Water Moderator Reactor," January 1965.

Review Status: Standard Nuclear Industry Code.

T30 WAPD-TM-678, W.R. Cadwell, "PDQ-7 Reference Manual," January 1978.

Review Status: Standard Nuclear Industry Code.

T31 WAPD-TM-743, J. B. Yasinsky, M. Natelson, and L.A. Hageman, "TWIGL - A Program to Solve the Two-Dimensional, Two Group, Space-Time Neutron Diffusion Equations with Temperature Feedback," February 1968.

Review Status: Standard Nuclear Industry Code.

T32 Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors (NRC Report) Pages 21 and 26.

Review Status: Approved.

T33 Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Federal Register, Vol. 39, No. 3, January 4, 1974.

Review Status: Approved.

T34 A Higher Ordered Difference Method for Diffusion Theory, C. P. Robinson & J. D. Eckhard, Trans. Am. Nucl. Soc., 15, Page 297, 1972.

Review Status: Standard Periodical Literature.

T35 Theory, Capabilities, and Use of the Three-Dimensional Reactor Operation and Control Simulator (ROCS), T. G. Ober, J.C. Stork, I.C. Rickard & J.K. Gasper, Nuclear Sciences & Engineering, 64, pp 605-623, 1977.

Review Status: Standard Periodical Literature.

14.3 METHODOLOGY CROSS INDEX FOR DBEs

DBE	References
RCS Depressurization	T2, T21, T22, T26*, T28
CEA Withdrawal	T2, T21, T22, T26*, T28
Loss of Coolant Flow	T1, T2, T11, T21, T22, T23*, T26
Full Length CEA Drop	T2, T21, T22, T26*
CEA Ejection	T25*, T29, T30, T31
Siezed Rotor	FSAR*, T2, T11, T21, T22, T30
Large Break LOCA	FSAR*, T3, T4, T5, T6, T7, T8, T9, T10,T11, T12, T13, T14, T17, T18
Small Break LOCA	T16*

*The asterisk indicates the reference in which the DBE and its associated analytical methodology is described. In all cases the event is described in the FSAR, but in those cases where the FSAR is not indicated a more complete description of the event and its associated analytical methodology is given in the reference topical indicated by an asterisk.