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NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. DPR-53

BALTIMORE GAS & ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-317

1.0 Introduction

By application dated February 23, 1979 and supplemental information dated January 12, February 7, March 5 and 13, May 7, 29 and 31, 1979, Baltimore Gas & Electric Company (BG&E or the licensee) requested an amendment to Facility Operating License No. DPR-53 for the Calvert Cliffs Nuclear Power Plant, Unit No. 1 (CCNPP-1). The amendment request consisted of:

- Technical Specification (TS) changes resulting from the analyses of Cycle 4 reload fuel;
- Approval to install a high burnup demonstration fuel assembly (SCOUT) and a prototype CEA; and
- Approval to operate another cycle with modified (sleeved and reduced flow) Control Element Assembly (CEA) guide tubes.

The associated specified TS changes are described in Section 4.0 of this Safety Evaluation (SE).

2.0 Background

In the Cycle 4 reload application for CCNPP-1 (Ref. 6), BG&E proposed to replace 40 Batch A and 32 Batch C fuel assemblies with 72 fresh Batch F fuel assemblies. The core related evaluations are presented in Sections 3.1 and 3.2 of this SE.

In December 1977, a severe CEA guide tube wear problem was identified at the Millstone Nuclear Power Station, Unit No. 2. Similar wear was subsequently found at CCNPP-1 and other facilities designed by Combustion Engineering (CE). The temporary repair for CCNPP-1 to allow Cycle 3 operation was to sleeve all fuel assemblies to be placed in CEA locations and the sleeving of other worn fuel assemblies in non-CEA locations to regain safety margins. Authorization for CCNPP-1 to operate for Cycle 3 in this mode was granted by Reference 1. As a result of the test program to evaluate the acceptability of the sleeves for a second cycle of operation, BG&E and CE found that some of the sleeves have become loose in the guide tubes (Ref. 14). The evaluation of the proposed repair and the entire CEA guide tube wear problem is presented in Section 3.3 of this SE.

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In the process of this review, we have requested and received additional information necessary for our evaluation (Refs. 10, 11).

CCNPP-1 is currently licensed to operate at 2700 MWt. The rated power level and all operating conditions remain the same for Cycle 4.

3.0 Evaluation

In this evaluation of a cycle reload for CCNPP-1, considerable use is made of generic reviews of various topical reports (See Topical References). Most of the topical reports have received formal NRC staff approval. In all cases where a topical report has not received approval, the report has been examined, its methods judged to be reasonable, and an appraisal has been made that a complete review will not reveal the methodology to be significantly in error. On this basis, all topicals referenced are judged to be acceptable for this reload evaluation.

3.1 Cycle 4 Fuel Design

The 217 fuel assembly Cycle 4 core will consist of:

BATCH IDENTIFICATION	WEIGHT & (w/o) ENRICHMENT	NUMBER OF FUEL ASSEMBLIES
B	#	1
D	#	48
D/	#	24
E	#	48
E/	#	24
F	3.03	48
F/	2.73	24

#Irradiated fuel from Cycle 3

As a result of the CEA guide tube wear problem, all fuel assemblies presently in Cycle 3 that will be placed in CEA locations in Cycle 4, with the exception of the Batch B test assembly and one other assembly, will have stainless steel sleeves installed in the CEA guide tubes in order to prevent guide tube wear. The Batch B test assembly was inspected during the current refueling outage and guide tube wear was found to be acceptable for another cycle of operation. The center core position occupied by the Batch B assembly is typically a low wear location for fuel assemblies. The other unsleeved fuel assembly in a CEA position is the result of a three way swap due to a problem sleeved fuel assembly as described in Reference 14. We find operation with two fuel assemblies unsleeved in CEA positions acceptable.

Of the new Cycle 4 fuel, eight Batch F assemblies and eight Batch F/ assemblies will be placed under dual CEAs and eight Batch F/ assemblies will be placed under single CEAs. These 24 new assemblies will have stainless steel sleeves installed in their CEA guide tubes.

BG&E has used the Cycle 3 reload analysis for CCNPP-1 as a "reference cycle" for the Cycle 4 reload analysis. Our original evaluation of Cycle 3 operation is presented in Reference 1. A reevaluation of Cycle 3 operation was necessary as a result of the reanalysis performed by BG&E in order to reach the licensed power level (Ref. 2). Analyses outside the envelope of the reference cycle have been reanalyzed.

3.1.1 Mechanical

In addition to the sleeving of fuel assemblies as described above and evaluated in Section 3.? of this SE, the following other changes have been made to the mechanical design of the new fuel assemblies.

Upper End Fitting Assembly - The holddown plate in the upper end fitting has been thickened slightly. Since this reduces the holddown spring working length, the free length of the springs has been reduced by the same amount. Therefore, the holddown force has remained constant.

Lower End Fitting - The cross-bracing which connects the lower end fitting posts has been thickened and raised 1/8" from the lowermost surface of the fuel assembly.

Guide Tube Flow Holes - 16 Batch F assemblies have guide tube flow holes identical in size to the Batch E fuel. Another 16 assemblies have the reduced flow holes described in Reference 6. This modification is identical to that made to 16 fuel assemblies installed in the present cycle at CCNPP-2 and evaluated in Reference 3. The remaining forty fuel assemblies were modified to have slightly less flow than the normal Batch E fuel assemblies.

The effect of the modified cooling flow through the CEA guide tubes on the thermal hydraulics of the core will be evaluated in Section 3.1.3 of this safety evaluation.

An analytical prediction of the time of cladding creep collapse for all Cycle 2 fuel has been performed by CE using the CEPAN code which has been reviewed and approved by NRC. From this analysis, it has been concluded by CE that the collapse resistance of all the fuel rods is sufficient to preclude cladding collapse during its design lifetime. The design lifetime of this fuel will not be exceeded during Cycle 4 operation. The Batch B fuel which is the most limiting with regard to clad collapse will have accumulated 35,400 Effective Full Power Hours (EFPH) by the end of cycle (EOC). This is below the predicted time to clad collapse which has been calculated to be greater than 38,500 EFPH for any standard fuel rod in this assembly. We have reviewed this analysis and found it to be acceptable.

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This cycle will also contain an additional change. This is the installation of a new fuel assembly called Scout which is a high burnup demonstration assembly that will provide information that will be useful in formulating a technical basis for the design, licensing and operation of fuel at high burnups for use in an extended fuel cycle.

The Scout high burnup demonstration assembly consists of 161 standard fuel rods and 15 demonstration rods. The mechanical design of the assembly components other than the 15 demonstration rods in this assembly is identical to the design of the other new fuel assemblies being loaded into the core. The 15 demonstration fuel pins are of two different mechanical designs. In one design, which is representative of six fuel pins, the spacer grid contacts the fuel pins at non-fueled regions. This could result in reduced grid/pin contact forces. To offset this possibility, the initial fill pressure in these rods was increased to decrease the magnitude of clad creepdown. A larger void volume exists in the rods with the greater initial pressurization which will result in no appreciable increase in the end of life internal pressure. CE has performed analytical predictions of the cladding creep collapse time for the demonstration fuel rods and has concluded that the collapse resistance of the demonstration fuel rods is sufficient to preclude collapse during their design lifetime. This lifetime will not be exceeded by the Cycle 4 duration.

3.1.2 Nuclear Analyses Methodology

The Nuclear Design Model used in previous cycles has been PDQ, a two-dimensional diffusion code using four energy groups. PDQ has been accepted industry wide. For Cycle 4, CE performed the calculations of certain parameters using the ROCS code instead of PDQ. Using a higher order differencing methodology than PDQ and only one and a half energy groups, ROCS is able to compute many parameters nearly as accurately as PDQ in three dimensions with more reasonable computer run time.

For Cycle 4, 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 banks
- Changes in 3-D core power distributions that result from inlet temperatures maldistributions (asymmetric steam generator transient)

None of these parameters require the detailed knowledge of pin powers normally computed by PDQ. BG&E states that in most cases, their parameters are calculated more accurately by ROCS because of its ability to account for three dimensional effects. BG&E has also stated that they observe guidelines to evaluate the adequacy of ROCS for computing these parameters on a case by case basis. If ROCS is judged to be not adequate for certain computation, then the computation is repeated using PDQ.

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

3.1.3 Nuclear Parameters

In the Reference 1 SE, we found that introducing of stainless steel sleeves into the CEA guide tube had minimal effect on reactor physics. The operation of the CCNPP-1 for one cycle with all CEA guide tubes sleeved has borne out this conclusion.

In the SE supporting the Cycle 2 reload for CCNPP-2 (Ref. 3), we approved a demonstration test consisting of 16 fuel assemblies with reduced CEA guide tube flow. BG&E has also proposed a 16 fuel assembly demonstration test for Unit 1 Cycle 4. They anticipate no substantial change in axial and radial power distribution as a result of the decreased flow in the modified CEA guide tubes. This demonstration test will be discussed in Section 3.3 of this SE.

The licensee has stated that 40 Batch F assemblies have a flow hole configuration that presents a greater flow area and a consequent increase in guide tube flow over the standard Batch E assemblies. Since the flow area is greater than the standard assemblies by only 4%, the licensee has judged this to have an insignificant effect on axial and radial power distributions.

The Batch F reload fuel is comprised of two sets of assemblies with two enrichments as previously described in Section 3.1 of this safety evaluation. Cycle 4 burnup is expected to be between 10,000 Megawatt Days per Metric Ton Uranium (MWD/MTU) and 10,555 MWD/MTU. The licensee has examined the Cycle 4 performance characteristics for a Cycle 3 termination point of between 8950 and 10,000 MWD/MTU. The actual Cycle 3 burnup, as stated by the licensee, was 9465 MWD/MTU.

The Cycle 4 moderator temperature coefficient is calculated to be $-0.4\times10^{-4}\,\Delta\rho/^\circ F$ at the EOC. The values for MTC are bounded by the values used in the reference cycle which are $-0.4\times10^{-4}\,\Delta\rho/^\circ F$ at beginning of cycle (BOC) and $-2.1\times10^{-4}\,\Delta\rho/^\circ F$ at EOC. We find these values of MTC to be acceptable.

Doppler coefficients calculated for Cycle 4 are $-1.50\times10^{-5}\,\Delta\rho/^\circ F$ at BOC hot zero power (HZP), $-1.20\times10^{-5}\,\Delta\rho/^\circ F$ at BOC hot full power (HFP) and $-1.37\times10^{-5}\,\Delta\rho/^\circ F$ at EOC HFP. These values are slightly more negative at HFP for both BOC and EOC conditions. Changes of this magnitude, 5% more negative at HFP BOC and 10% more negative at HFP EOC have a minimal impact on the analysis of postulated Anticipated Operational Occurrences (AOOs) and accidents that result in a reactor cooldown. The slightly more negative values of the Doppler coefficient act to add additional conservatism to AOOs and accidents during which fuel temperature is tending to increase. We find the values of the Doppler coefficient calculated for Cycle 4 to be acceptable.

The total delayed neutron fraction for Cycle 4 has decreased slightly at EOC and increased slightly at BOC from that in the reference cycle. This would have a minor impact on the CEA ejection accident. The CEA ejection accident has been reanalyzed and is discussed in Section 2.5 of this safety evaluation.

At EOC 4, the reactivity worth of all CEAs inserted, less the highest worth CEA stuck allowance, is 7.7%$\Delta\rho$. The reactivity worth required to shut down the plant including power defect HFP to HZP, shutdown margin and safeguards allowance required to control the steam line break incident at EOC 4 is 6.2%$\Delta\rho$. The margin available in negative reactivity is 1.5%$\Delta\rho$ which is more than adequate to account for any uncertainty in nuclear calculations. We find these shutdown margins to be acceptable.

3.1.4 Thermal Hydraulics

The licensee states that the steady state Departure from Nucleate Boiling Ratio (DNBR) analyses of Cycle 4 at the rated power of 2700 MWT/MWt has been performed using the TORC code which employs the CE-1 DNBR correlation. The TORC code has been approved by Reference h for use in licensing and the CE-1 correlation has been approved with a 1.19 DNBR limit. TORC/CE-1 was also used in the generation of limiting conditions for operation (LCOs) on DNBR margin in the TS and all AOOs and postulated accidents which were reanalyzed for Cycle 4.

The fuel rod bowing effects on DNB margin for CCNPP-1 have been evaluated within the guidelines set forth in Reference g, as approved in the reference cycle SE (Ref. 1).

A total of 81 fuel assemblies will exceed the NRC-specified DNB penalty threshold burnup of 24,000 MWD/MTU, as established in Reference g, during Cycle 4. At the end of Cycle 4, the maximum burnup attained by any of these assemblies will be 42,800 MWD/MTU. From Reference g, the corresponding DNBR penalty for 42,800 MWD/MTU is 6.30 percent.

An examination of power distributions for Cycle 4 shows that the maximum radial peak at hot full power in any of the assemblies that eventually exceed 24,000 MWD/MTU is at least 10.30% less than the maximum radial peak in the entire core. Since the percent increase in DNBR has been confirmed to be never less than the percent decrease in radial peak, there exists at least 10.30% DNBR margin for assemblies exceeding 24,000 MWD/MTU relative to the DNBR limits established by other assemblies in the core. This margin is considerably greater than the Reference f reduction penalty of 6.30% imposed upon fuel assemblies exceeding 24,000 MWD/MTU in Cycle 4. Therefore, no power penalty for fuel rod bowing is required in Cycle 4.

The modifications to the fuel assemblies to alleviate the CEA guide tube wear problem have a small effect on their thermal hydraulic performance. As identified previously in this SE, Cycle 4 will have essentially two different modifications: 1) guide tube sleeving and 2) reduction in guide tube flow.

The flow characteristics of the assemblies with four 0.25" diameter hole and one 0.125" diameter hole and the assemblies with four 0.25" diameter holes and three 0.093" diameter holes are essentially equivalent.

The guide tube sleeving affects thermal hydraulic performance in three areas: core bypass flow, boiling in the guide tube sleeve annulus, and CEA cooling. As stated by the licensee, sleeving reduces the guide tube flow from 1400 lbm/hr to 700 lbm/hr. This change, however, compared to total core bypass flow is a minor effect which is in the conservative direction; i.e., it tends to increase the flow slightly through the core. Bypass flow must be maintained below 3.7% to preserve the design thermal margin. Sleeving improves this margin.

The second area of consideration is the potential for boiling in the guide tube sleeve annulus. The licensee states that no boiling will occur in the region in which the sleeve is expanded into contact with the guide tube since the CEA linear heat rate of 3.68 KW/ft is below the boiling limit of 6.5 KW/ft. In the non-expanded region, axial peaks can be maintained such that CEA linear heat rates are below the 1.2 Kw/ft boiling limit. Therefore, boiling is unlikely in this region. If boiling does occur, slots and holes in the sleeve assure that any expansion due to boiling is relieved and no mechanical damage will be caused. It is our opinion that limited boiling in this region is acceptable.

The criteria for adequate CEA cooling is that there is no bulk boiling in the guide tube during operation. The licensee states that cooling flow of 388 lbm/hr is required to meet this criteria. The cooling flow of 700 lbm/hr exceeds the minimum by a substantial margin. We find this to be acceptable.

The 16 fuel assemblies will have reduced guide tube cooling flow due to the reduction in number and size of the flow holes. The CEA cooling flow for this design has been stated by the licensee to be 565 lbm/hr. This exceeds the bulk boiling criteria of 388 lbm/hr and has a minimal impact in the conservative direction on total core bypass flow. However, for Cycle 4 none of these 16 assemblies will be in CEA locations.

The licensee has stated that the maximum peaking factor in any fuel rod in the Scout high burnup demonstration bundle is predicted to be more than 12% below the limiting pin peak in the core and the maximum pin peaking factor in any demonstration rod is predicted to be more than

15% below the limiting pin peak in the core. Considering that the bundle geometry of the Scout assembly is identical to the other Batch F assemblies and the Scout assembly power is well below the limiting core bundle the thermal hydraulic design of this assembly is acceptable.

3.2 Uncertainty in Nuclear Power Peaking Factors

In-core detector measurements are used to compute the core peaking factors using the INCA Code (Ref. c). The coefficients required to perform this data reduction are performed using the methodology described in the topical report.

For Cycle 4 operation, the licensee has proposed measurement uncertainties of 6% for the total integrated radial peaking factor (F_r) and 7% for the total power peaking factor (F_q) for base load operation and 8.0% and 10.0% for load follow operation.

The initial CE evaluation of peaking factor uncertainty was presented in References c and d. In a meeting with CE on March 6, 1979, data was presented showing measurement uncertainty of 6% in Fr and 7% in Fq to be conservative (Ref. 8). On this basis, we find these measurement uncertainties of 6% and 7% for Fr and Fq, respectively, to be acceptable without the load follow operation restrictions.

3.3 CEA Guide Tube Integrity

BG&E instituted an Eddy Current Testing (ECT) inspection program at CCNPP-1 to ascertain the condition of sleeves in assemblies located under CEA's during Cycle 3 (Ref. 4). No indications of sleeve wear were found in these assemblies, however several guide tube sleeves, when subjected to pull tests, did not exhibit the expected resistance to axial motion (Ref. 14). Because the CCNPP-1 wear inspection program showed ECT signals with widely varying magnitudes at the crimped regions of the sleeves, the inspection program was extended to assess the crimp size in a number of different type fuel assemblies. This inspection for crimp integrity was performed using the same probe and test procedure used in the wear inspection program.

The results of these inspections revealed a large number of sleeved fuel assemblies outside the ECT and pull test acceptance criteria used at other CE designed facilities. The explanations of CCNPP-1 results in comparison with the results from the other CE facilities were that the sleeving sequence used at other facilities in 1978 differed from that used at CCNPP-1 (the first facility where sleeving was performed). At the other facilities, pull tests were performed on the sleeves after the crimping step to verify the adequacy of the crimp. Following the "crimp verification" pull test, expanding steps were then performed on the sleeves. However, at CCNPP-1, the pull tests were not performed until after both the crimping and the expanding steps were completed. The licensees and CE have concluded that this sequence change added frictional resistance between the expanded sleeve and the guide tube wall to mask the presence of inadequate crimps that would have been identified by an intermediate "crimp verification" pull test.

In addition, the low ECT results at CCNPP-1, which indicate inadequate crimps, were unique to a particular fuel category. This fuel category consists of those assemblies that had been irradiated prior to sleeving in 1978. In this fuel category at CCNPP-1, the EC signals were low for approximately 50% of the 235 sleeves tested. The low signals for irradiated fuel were not evident at the other facilities. Thus, it appears that the increased yield strength of irradiated guide tubes reduced the displacement of the crimp.

To remedy the observed inadequacy of the crimps at CCNPP-1, a total of 28 assemblies were designated for recrimping, using the new style crimp over the previously made old style crimp. ECT was performed on each sleeve after recrimping to measure actual crimp size. The basis of selecting the 28 fuel assemblies was that these assemblies were in the category of those assemblies sleeved in 1978 in the irradiated condition and are to be under CEAs for Cycle 4 operation. Because the recrimp is positioned at some distance from the bottom of the sleeve, a second operation, in which the bottom is re-expanded against the guide tube wall, was also performed. This operation, together with a free path gauge check was used to insure that the end of the sleeve would not interfere with CEA insertion.

The licensee stated that bench tests were completed on sample guide tube and sleeves to determine effects on sleeve and guide tube geometry by installing a second crimp over a previously installed crimp. Results of these test samples showed that the new style crimp can be installed over the old style crimp without "rolling in" the end of the sleeve, or causing any other anomalies in geometry. The tests also indicated no need for an additional lower end expansion; however, this procedure was retained in field crimping operations to preclude any chance of sleeve edge protrusion. For the actual recrimps placed in the fuel assemblies in question, all sleeves have been ECT and shown to have crimp sizes sufficient to prevent axial motion (Ref. 14).

All other crimping and sleeving operations for this outage have used the new style crimping tools. The higher crimp pressure inherent with the new style crimp provides a greater force to locally deform (crimp) the higher strength irradiated guide tubes and likewise provides a more defined crimp geometry to resist axial motion of the sleeves.

We have reviewed the proposed crimping, and recrimping of the CEA guide tubes. and the results of the surveillance tests at CCNPP-1. Based on the information provided in Reference 14, we agree that the guide tube sleeving operations at CCNPP-1 provide acceptable repairs to the guide tubes for Cycle 4 operation.

In Reference 14, BG&E stated that CE recommended operational guidelines to reduce relaxation effects in the guide tube sleeves during Cycle 4 operation. This recommended guideline is to restrict movement of the CEAs at systems temperatures below 400 F except for normal movement associated with refueling operations. We find the recommended operational guideline reasonable. BG&E has agreed to implement this restriction on CEA movement.

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Sixteen Batch F fuel assemblies have been modified by decreasing the number and size of the flow holes and the size of the bleed holes. Tests have indicated that the resulting decrease in guide tube flow was accompanied by less CEA flow-induced vibration and, therefore, less guide tube wear. The SE for CCNPP-2, Reference 3, found the demonstration test similar to that proposed for CCNPP-1 with 16 fuel assemblies to be acceptable. The increase in the CEA insertion time to 3.1 seconds was also found acceptable. We, therefore, conclude that the demonstration test of 16 modified fuel assemblies with reduced guide tube flow is acceptable for Cycle 4 operation of CCNPP-1.

BG&E has agreed to provide a Cycle 5 guide tube evaluation program, identifying changes from the Cycle 4 program at least 90 days prior to the CCNPP-1 shutdown for the Cycle 5 reload outage.

3.4 Analyses of Anticipated Operational Occurrences (AOOs)

Reference 5 discusses the safety analyses of postulated AOOs for CCNPP-1 Cycle 4. The licensee classifies the list of postulated AOOs into two categories. The first category includes those AOOs for which the Reactor Protection System (RPS) Limiting Safety System Settings (LSSS) as specified in the plant TS assure that the Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded. The second category includes those AOOs for which initial steady state overpower margins are maintained by adherence to the Limiting Conditions for Operation (LCOs) specified by the TS for the plant. Adherence to the LCOs assure that SAFDL limits are not exceeded.

The loss of flow transient causes the most rapid change in DNBR and both a reactor trip and steady-state overpower margin is required to maintain the SAFDLs. The LCOs and LSSSs for Cycle 4 TS were calculated using the methods described in Reference f. The required AOO reanalyses were done using the computer code CESEC (Ref. i).

The licensee stated in Reference 5 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 in case it could lead to a significant relaxation of TS.

The results of that comparison show that the key parameters to all the AOOs and postulated accidents for Cycle 4 operation are the same as the specified reference cycle input parameters, except for the following:

1. CEA drop time to 90% inserted
2. Integrated radial peaking factor (F_r)
3. Seized rotor pin census
4. Core bypass flow fraction
5. RTD response time

For all AOOs and postulated accidents other than those reanalyzed, the licensee has stated that the CCNPP-1 safety analysis submitted either in the FSAR or in previous reload cycle license submittals bound the results that would be obtained for Cycle 4 and demonstrate continued safe operation of CCNPP-1 at 2700 MWt.

Since the CEA drop time to 90% insertion has increased for Cycle 4, the Loss of Flow Event, CEA Ejection Event, RCS Depressurization Event, Seized Rotor Event and the CEA Withdrawal Event were reanalyzed. These events are adversely impacted by the CEA drop time, since a reactor trip is necessary to terminate the event.

The sleeving of the CEA guide tubes has a negligible effect on CEA rod drop times but the reduction of the CEA guide tube flow holes does impact on the rod drop times. As previously stated, the Cycle 4 reload will have 16 fuel assemblies with reduced flow holes. The effect of these flow holes on rod drop times is to increase the time to 90% insertion from 2.5 to 3.1 seconds. BG&E has identified this as a proposed change to the TS 3.1.3.4 at this time, even though none of these assemblies are under CEAs during this cycle. To assess the impact of this change in rod drop time, the licensee has examined all the design basis events which could require a trip to prevent exceeding SAFDL limits. An evaluation of these design basis events showed that only five events may be adversely affected by increased scram time. For these evaluations, it was conservatively assumed that all the CEAs are inserted at the same insertion versus time characteristic curve as in the 16 fuel assemblies with the reduced guide tube flow. Those transients which were reanalyzed are discussed below.

BG&E has proposed a change to the TS Table 2.2-1 raising the high power level trip from 106.5% to 107.0% power. The safety analysis assumes a trip at 112% of rated power. A 5% power measurement uncertainty has always been applied in the process of generation LSSS limits. In the past, this uncertainty was applied in a multiplicative fashion (which yields the equivalent of a 5.5% of rated power uncertainty), but evaluations showed that application of the uncertainty in this fashion is conservative. In accordance with current methods (as described in Reference f), the power measurement uncertainty is now deducted algebraically. It is this difference in the manner in which the uncertainty is applied that leads to the 107% versus 106.5% LSSS limit. We have reviewed this change and find it to be acceptable.

3.4.1 CEA Withdrawal Event

The CEA Withdrawal event was reanalyzed for Cycle 4 due to the increase in the Resistance Temperature Detector (RTD) response time to envelope future cycles and the increase in the CEA drop time to 90% insertion from 2.5 seconds to 3.1 seconds. The CEA Withdrawal event was reanalyzed for reactor initial conditions of zero power and full power and the licensee has stated that the Departure from Nucleate Boiling (DNB) and fuel centerline melt Specified Acceptable Fuel Design Limits (SAFDLs) will not be exceeded during CEA Withdrawal transient.

The CEA Withdrawal transient initiated at rated thermal power results in the maximum pressure bias factor of 62.0 psia. This bias factor accounts for measurement system processing delays during the CEA Withdrawal event. The pressure bias factor for this cycle has increased from the reference cycle due to the increase in the RTD time constant and the increase in the CEA drop time to 90% insertion. This pressure bias factor is used in generating TM/LP trip setpoints to prevent the SAFDLs from being exceeded during a CEA Withdrawal Event. The TS have been changed to reflect the 62.0 psia pressure bias factor. We find this analysis and the change to the plant TS to be acceptable.

3.4.2 RCS Depressurization Event

The RCS Depressurization event was reanalyzed for Cycle 4 to assess the impact of increasing the CEA drop time to 90% insertion from 2.5 seconds for Cycle 3 to 3.1 seconds for Cycle 4. As stated in Reference f, this is one of the events analyzed to determine a bias term input to the TM/LP trip. Hence, this event was analyzed for Cycle 4 to obtain a pressure bias factor. This bias factor accounts for measurement system processing delays during this event. The trip setpoints incorporating a bias factor at least this large will provide adequate protection to prevent the DNBR SAFDL from being exceeded during this event.

The analysis of this event shows that the pressure bias factor is 35 psia which is less than that required by the CEA Withdrawal Event. Hence, the use of the pressure bias factor determined by the CEA Withdrawal event will prevent exceeding the SAFDLs during an RCS Depressurization event.

3.4.3 Loss of Coolant Flow Event

The Loss of Coolant Flow event was reanalyzed for Cycle 4 to determine the impact on margin requirements that must be built into the LCOs due to the increase in the CEA drop time to 90% insertion.

The low flow trip setpoint is reached at 1.0 seconds and the CEAs start dropping into the core one second later. A minimum DNBR of 1.25 is reached at 2.3 seconds.

The low flow trip, in conjunction with the initial overpower margin maintained by the LCOs in the TS assure that the minimum DNBR will be greater than or equal to 1.19 for the Loss of Coolant Flow Event.

3.4.4 Conclusion

We have reviewed the licensee's analyses of AOOs for Cycle 4 operation of CCNPP-1 and conclude that they are acceptable.

3.5 Postulated Accidents Other Than LOCA

The licensee has reviewed the postulated accidents other than LOCA. Reference 5 discusses the safety analysis performed for this category of accident for CCNPP-1 Cycle 4. Postulated accidents as other plant events, need to be reanalyzed only if the key parameters influencing the event are not enveloped by the reference cycle data. Those accidents that were reanalyzed are discussed below.

3.5.1 CEA Ejection Event

The CEA Ejection Event was reanalyzed for Cycle 4 to assess the impact of increasing the CEA drop time to 90% insertion and the increase in the augmentation factor in comparison to the reference cycle. In addition, the zero power case was analyzed due to the decrease in axial peak in comparison to the reference cycle. The reference cycle for this event is the analysis upon which the licensing of CCNPP-2 Cycle 2 was based. Our evaluation of this reload is found in Reference 3. Hence, this event was reanalyzed to demonstrate that the criterion for clad damage is not exceeded during Cycle 4 operation.

The licensee's analysis shows that for both the zero power and full power cases the clad damage pellet enthalpy threshold of 200 cal/gm is not violated. Therefore, no fuel rods are predicted to suffer clad damage.

3.5.2 Seized Rotor Event

The Seized Rotor event was reanalyzed for Cycle 4 due to the changes in the following key parameters.

- The increase in the CEA drop time to 90% insertion
- The decrease in core bypass flow, which increases the net core flow
- The decrease in the Radial Peaking Factor
- A more adverse (flatter) pin census.

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The increase in the CEA drop time and the flatter pin census adversely impact the consequences of this event. Increasing the net core flow and decreasing the Radial Peaking Factor will decrease the consequences of this event. Hence, a reanalysis was performed for Cycle 4 to ensure that only a small fraction of fuel pins are predicted to fail during a Seized Rotor event.

A conservatively "flat" pin census distribution (a histogram of the number of pins with radial peaks in intervals of 0.1 in radial peak normalized to the maximum peak) was used to determine the number of pins that experience DNB.

The results indicate that increasing the core flow and decreasing the radial peaking factor offset the increase in the CEA drop time to 90% insertion. It was calculated that for Cycle 4, less than 0.5% of fuel pins will experience DNB for even a short period of time.

For the case of the loss of coolant flow arising from a seized rotor shaft, it is assumed that there is an instantaneous reduction to three pump flow. The low flow trip assures that less than 0.5% of fuel pins experience DNB. This is the same as that calculated for the reference cycle. Hence, the conclusions reached for reference cycle remain valid for Cycle 4.

3.5.3 Conclusions

We have reviewed the accident analyses for events other than LOCA for CCNPP-1 Cycle 4 and conclude that they are acceptable.

3.6 Cycle 4 LOCA Analysis

Reference 5 provides a comparison of the fuel specific parameters for the limiting fuels during Cycles 3 and 4.

The Cycle 4 core contains 216 high density fuel assemblies and one low density Batch B assembly. The highest power pin in the low density Batch B assembly will not achieve a power level greater than 75% of the highest power pin in the core. Therefore, a Batch B fuel pin will not be limiting in Cycle 4.

The remaining 216 high density fuel assemblies contain 72 partially depleted Batch D assemblies, 72 partially depleted Batch E assemblies and 72 fresh Batch F assemblies. Burnup dependent calculations were performed for the high density fuel assemblies with the FATES (Ref. b) and STRIKIN-II(Ref.a) codes. The results demonstrate that the most limiting fuel pin during Cycle 4 is located in one of the partially depleted Batch E assemblies.

The limiting high density fuel in Cycle 4 has a stored energy 268°F lower than the limiting fuel in Cycle 3. Consequently, the ECCS performance results reported for Cycle 3 conservatively bound the performance for Cycle 4. Therefore, the peak linear heat generation rate of 14.2 KW/ft which was demonstrated to be acceptable for Cycle 3 is also an acceptable limit for Cycle 4 operation.

In order to comply with 10 CFR 50, Appendix K, the LOCA analysis must demonstrate that the peak clad temperature (PCT) remains below 2,200 F and the maximum local cladding oxidation, which is a function of the time dependence of the PCT, remains below 17 percent.

During a LOCA, the cladding swells due to the decreased coolant pressure and the increased fuel temperature and gas pressure. The clad swelling is terminated if the cladding ruptures. 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 occurs during the reflood phase of the LOCA, the Rupture-Strain curve used by CE is possibly nonconservative. However, this is not a problem for CCNPP-1,because clad rupture is predicted to occur during the blowdown phase and not the reflood phase. The staff review has found the CE analyses for the case of rupture during the blowdown phase to be acceptable.

We conclude, as a result of our review, that the CCNPP-1 Cycle 4 ECCS performance is in conformance with the criteria specified in 10 CFR 50.46(b) and is, therefore, acceptable.

4.0 Technical Specifications

The TS changes proposed for this amendment are summarized in the following statements.

Page 1-3

The definition of Shutdown Margin (Section 1.13) would be revised to eliminate the reference to part length CEAs.

Page 2-7

The Power Level-High RPS trip would be increased 0.5% to 107.0% as a result of the Cycle 4 analyses.

Pages 2-12 & 2-13

Figures 2.2-2 and 2.2-3, relating to the TM/LP trip setpoint, would be modified as a result of the Cycle 4 analyses.

Page 3/4 1-23

The CEA drop time, TS 3.1.3.4, would be increased from 2.5 seconds to 3.1 seconds as a result of the changed hydraulic characteristics of the 16 demonstration fuel assemblies.

Pages 3/4 2-4 & 3/4 2-5

New axial flux offset (Figure 3.2-2) and augmentation factors (Figure 4.2-1) would be added based on revised physics calculations.

Pages 3/4 2-8 & 3/4 2-9

These power distribution limit changes would be made based on revised physics calculations and application of the standard CE setpoint methodology.

Page 3/4 2-11

Figure 3.2-4 would include the increase in allowable azimuthal tilt.

Page 3/4 2-13

The old TS 3.2.5 would be eliminated since the core can not achieve a core exposure that would result in clad collapse.

Page 3/4 2-15

Table 3.2-1 would be revised to increase the cold leg temperature used in DNB calculations by 1 F to 548 F. Parameter values for less than four RCP operation would be eliminated pending NRC review of ECCS analyses for operation in that mode.

Page 3/4 3-6

Table 3.3-2 would be revised to increase the RTD response time from 5 to 8 seconds in accordance with the Cycle 4 analysis.

5.0 Physics Startup Testing

The physics startup test program as described in Reference 6 has been reviewed. The low power tests include CEA symmetry check, critical boron concentration measurements, isothermal temperature coefficient measurements and CEA group worth measurements. The power ascension tests include power coefficient and power distribution tests.

The staff discussed the CEA symmetry test and the review criteria for this test with the licensee. The licensee agreed to perform the CEA symmetry test on 2 shutdown banks and review criteria as stated in Reference 13. The review criteria for power distribution measurements are also given in Reference 13.

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The staff finds the entire program including the acceptance and review criteria and the remedial actions acceptable.

6.0 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.

7.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of 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: June 14, 1979

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b. CENPD-139, "CE Fuel Evaluation Model", July 1974.

c. CENPD-145, "INCA: Method of Analyzing In-Core Detector Data in Power Reactors", April 1975.

d. CENPD-153, "Evaluation of Uncertainty in the Nuclear Form Factor Measured by Self-Powered Fixed In-Core Detector Systems", August 1974.

e. CENPD-161-P, "TORC Code - A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975.

f. CENPD-199-P, "CE Setpoint Methodology", April 1976.

g. CENPD-225, "Fuel and Poison Rod Bowing", October 1976.

h. Evaluation of Topical Report CENPD-161-P, K. Kniel (NRC) to A. E. Scherer (CE), September 14, 1976.

i. CENPD-107, "CESEC-Digital Simulation of a CE Nuclear Steam Supply System," April 1974.

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2. NRC Amendment No. 33 for CCNPP-1, Cycle 3 Reanalysis, R. W. Reid to A. E. Lundvall, June 30, 1978.

3. NRC Amendment No. 18 for CCNPP-2, Cycle 2 Reload, R. W. Reid to A. E. Lundvall, October 21, 1978.

4. BG&E Sleeved CEA Guide Tube Inspection Program, A. E. Lundvall to R. W. Reid, January 12, 1979.

5. BG&E High Burnup Demonstration Program, A. E. Lundvall to R. W. Reid, February 7, 1979.

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10. NRC Request for Additional Information, R. W. Reid to A. E. Lundvall, April 13, 1979.

11. BG&E Reponse to Cycle 4 Reload Questions, A. E. Lundvall to R. W. Reid, May 7, 1979.

12. NRC Safety Evaluation - Small Break LOCA Analysis with No Credit for Charging Pump Flow, R. W. Reid to A. E. Lundvall, May 18, 1979.

13. BG&E Supplement 4 to Application for Cycle 4 Reload - Physics Startup Testing, A. E. Lundvall to R. W. Reid, May 29, 1979.

14. BG&E Supplement 5 to Application for Cycle 4 Reload - CEA Guide Tube Test Results and Repairs, A. E. Lundvall to R. W. Reid, May 31, 1979.