UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 115 TO LICENSE NO. DPR-49

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 Introduction

The Duane Arnold Energy Center (DAEC) was designed and constructed to operate at a steady state core power of 1658 MWt. The staff reviewed the Final Safety Analysis Report (FSAR) and the Environmental Report (ER) and issued a Safety Evaluation Report (SER) and a Final Environmental Statement (FES), addressing the operating power level of 1658 MWt. The staff stated in the SER that the DAEC power be restricted to 1593 MWt until the licensee satisfactorily resolved the power ascension program issues. Accordingly, the DAEC license was issued for a maximum power level of 1658 MWt but the rated power was restricted in the Technical Specifications to 1593 MWt. The Iowa Electric Light and Power Company (licensee), by letter dated August 17, 1984, proposed a revised power ascension program consistent with the staff recommendations in the DAEC SER, and requested an increase in their rated power from 1593 MWt to 1658 MWt. Since the issuance of the DAEC license in February 1974, several changes have occurred in the regulations and the regulatory bases. The Commission has since issued the Appendix K to 10 CFR 50, outlining the calculational models to be used to satisfy the Emergency Core Cooling System (ECCS) requirements. The Appendix K requires that the loss of coolant accident (LOCA) analyses be performed at 102% of the maximum authorized power level. Additionally, the staff's review philosophy outlined in the Standard Review Plan (SRP) was revised in 1980. Responding to these changes, the licensee has redone the LOCA analyses for a power level two percent above the proposed maximum power of 1658 MWt. The licensee has also evaluated the impact of all the changes on the reactor systems performance, core performance, engineered safety features, and the design basis accidents and their consequences. All evaluations have been done using the guidance contained in the revised Standard Review Plan.

2.0 Evaluation

The staff reviewed the licensee's application and the supporting analyses and evaluations. The staff's evaluations are summarized as follows:

PLANT HEAT BALANCE

To determine the necessary plant initial conditions and input parameters, heat balances at the proposed power level of 1658 MWt and 1691 MWt were performed. Heat balance parameters are used as initial conditions and input parameters in various plant analyses-LOCA, thermal limits, reactor overpressure protection, reactor internal pressure differences and containment evaluation.

An increase in thermal power from 1593 MWt to 1658 MWt will require an increase in the reactor nominal operating pressure from 1005 psig to 1025 psig. This increase in pressure will compensate for the additional pressure drop caused by the increased steam flow to the turbine and will provide sufficient control margin for the turbine control valves so that continuous stable operation will be maintained.

This new reactor dome pressure will affect reactor systems which are pressure dependent. Reactor high pressure scram setpoint, ATWS reactor recirculation pump trip and SRV setpoints are affected and will be addressed as the Technical Specification Changes later.

LOSS-OF-COOLANT ACCIDENT

The LOCA analyses were performed using Appendix K approved ECCS evaluation models/methodology to demonstrate conformance with the ECCS acceptance criteria of 10 CFR 50.46 at uprated power conditions. The ECCS performance was evaluated for the entire LOCA break spectrum. The ECCS performance evaluation included the most limiting break size, break location, and single failure combinations. The LOCA analysis results improved from the previous analyses due to credit taken for a full-core of Drilled Lower Tie Plates which permit a substantial amount of backflow leakage over the range of differential pressures expected. The licensee has demonstrated compliance with the ECCS acceptance criteria as follows.

(1) Peak Cladding Temperature (PCT)	1959°F	(2200°F allowable)
(2) Maximum cladding oxidation	1.1%	(17% allowable)
(3) Maximum total hydrogen generation	0.08%	(1.0% allowable)

(Recirculation Suction Line Break-LPCI Injection Valve Failure)

(4) A coolable geometry is demonstrated by the compliance with the criteria for the PCT and the maximum cladding oxidation.

(5) Long-term cooling is ensured by the use of redundant systems that have adequate water sources available to remove the decay heat generated within the reactor core and transfer the heat to the ultimate heat sink.

The results of the LOCA analyses demonstrate that the ECCS will perform its function in an acceptable manner and meet the 10 CFR 50.46 acceptance criteria.

REACTOR VESSEL OVERPRESSURE PROTECTION

The DAEC pressure relief system includes two spring safety valves and six dual function safety/relief valves. The pressure relief system was designed in compliance with ASME Code Section III. Article NB-7000 which requires that the maximum pressure reached during the most severe pressure transient be less than the 110% of the reactor vessel design pressure. This pressure limit is 1375 psig.

The two spring safety valves are set to actuate at 1240 psig. Due to the increase in dome pressure of 10 psi, the proposed setpoints for the six dual function SRVs are 1110, 1120, 1130 (two valves) and 1140 (two valves) psig. The DAEC nominal operating pressure is 1025 psig.

The MSIV closure with flux scram (assuming position switch scram failure) transient was analyzed to determine the peak system pressure. The SRVs open to limit the pressure rise at the bottom of the vessel to 1275 psig which is below the allowable maximum pressure of 1375 psig.

The DAEC pressure relief system has adequate simmer and overpressure protection margin during plant operation at uprated power conditions.

ABNORMAL OPERATIONAL TRANSIENTS AND MCPR OPERATING LIMITS

To maintain fuel cladding integrity, the reactor core is designed with appropriate margin during any conditions of normal operation including the effects of anticipated operating occurrences. The minimum value of the critical power ratio reached during the transient should be such that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. The limiting value of the minimum critical power ratio is called the safety limit.

The design calculation of the safety limit MCPR is based on a Monte Carlo analysis (ODYN) of core performance in a limiting configuration and takes into account both performance monitoring uncertainties and calculation uncertainties. The safety limit MCPR is 1.07 for the DAEC.

To determine the MCPR operating limit for the DAEC at the uprated conditions, the most limiting abnormal operational transients were considered in the analysis.

These transients are:

(1) Load Rejection without Bypass
(2) Turbine Trip without Bypass
(3) Loss of Feedwater Heating
(4) Feedwater Controller Failure
(5) Inadvertent Startup of HPCI Pump

The DAEC uprated MCPR operating limit is obtained by addition of the absolute maximum CPR value (including any imposed adjustment factors) for the most limiting transient postulated to occur at the plant from uprated conditions, to the fuel cladding integrity safety limit.

The ODYN Option A MCPR operating limit for the DAEC, Cycle 8, is 1.28 and the ODYN Option B MCPR operating limit is 1.26.

The MCPR operating limit ensures that the fuel cladding integrity would be maintained for any abnormal operational transient.

SRV LOW-LOW-SET SYSTEM (LLS)

The LLS logic system employs two non-ADS SRVs to reduce subsequent actuations of SRVs during plant abnormal transients or small break LOCAs. The LLS system would mitigate the induced thrust loads on the SRV discharge line resulting from SRVs subsequent actuations.

For the DAEC uprated power, the two limiting events, (1) Isolation by MSIV closure and (2) Small Break with Isolation Due to Loss of Offsite Power were analyzed. The results of the analysis indicate that the minimum time between SRV actuation for both events exceeds the minimum acceptable value of 3.7 seconds mitigating thrust loads.

We conclude that there is no adverse effect on the LLS logic system due to power uprate.

FUEL MECHANICAL DESIGN

The fuel used in the DAEC is the standard General Electric design which has been described in the GESTAR document (Reference 3). This fuel design has been approved for use in BWR reactors from BWR/2 through BWR/6 designs with power densities which encompass that of DAEC at the uprated power. We therefore find its use acceptable. The safety limits for the fuel

(MCPR and clad strain limits) are established generically as described in Reference 3. This report has been reviewed by the staff and approved (Reference 4). We conclude that these limits apply to DAEC with uprated power.

NUCLEAR DESIGN

The methods and techniques employed in the nuclear design of DAEC at uprated power are described in Reference 3. These methods have been approved by the staff for use in the design and analysis of BWR cores, including those having core power densities in the range of uprated DAEC. We conclude that the nuclear design of the uprated core is acceptable.

THERMAL-HYDRAULIC DESIGN

The methods and techniques used to perform the thermal-hydraulic design of the uprated DAEC core are described in Reference 3 which has been approved by the staff for such application.

The value of 1.07 for the MCPR safety limit is a generic value applicable to reloaded BWR reactors having 8x8, 8x8R, P8x8R, and/or BP8x8R fuel. The methods and techniques used to obtain the safety limit value are described in Reference 3. The value of the safety limit MCPR depends on uncertainties in the thermal hydraulic parameters of the core and on the uncertainty in the critical heat flux correlation (GEXL). Since these quantities are not affected by the power uprate, we conclude that the safety limit MCPR value of 1.07 is still acceptable for DAEC.

The operating limit MCPR is obtained by an analysis of the transients to obtain their effect on the core critical power ratio (ΔCPR). The maximum value of ΔCPR is then added to the safety limit MCPR to obtain the operating limit. The maximum value of ΔCPR includes multipliers which are required to account for uncertainties in the transient calculation methods.

The methods and techniques employed in obtaining the operating limit MCPR are described in Reference 3. These methods are applicable generically to BWR reload analysis. We find that they are acceptable for the uprated DAEC.

The K_f curves are plots of multiplying factors to be applied to the operating limit MCPR for core conditions less than rated flow. The rated flow is the same for the uprated as for the present DAEC. However, the steam flow will be greater by a factor of $\sim$1.048. Since the current K_f curves were calculated for 105 percent of the rated steam flow, the current values of the K_f factors are still applicable.

The thermal-hydraulic stability analysis of the DAEC core has been performed at the proposed uprated power level for previous cycles as well as for Cycle 8. Therefore the power uprate does not affect the analysis. The effects of single loop operation on core thermal-hydraulic performance will be addressed in a separate evaluation.

TRANSIENTS AND ACCIDENTS

Transient and accident analysis methods are described in Reference 3. These are the same methods that have been used in previous cycles and they continue to be acceptable. The applicability of these methods is not dependent on reactor power within the range of BWR designs. Accordingly, the transient and accident analysis are acceptable at the DAEC uprated power.

EXTENDED LOAD LINE LIMIT ANALYSIS

The operation with an extended load line would permit higher power at low flows than the power permitted by operation with the standard load line. This affects the core thermal hydraulic stability analysis and the initial conditions for certain transients. The power uprate will have an additional effect on these parameters. The effect of the power uprate on the Extended Load Line Analysis will be treated in the review of that analysis which will be the subject of a separate evaluation.

CONTAINMENT RESPONSE

The containment design basis accident (DBA) is an instantaneous double-ended guillotine break of the recirculation pump suction line which is postulated to occur. Analyses to determine the DAEC containment short-term accident response were performed at an initial power condition corresponding to 102% of the uprated power. The differences in the peak calculated values for the drywell and wetwell pressures and temperatures between the uprated power conditions of 102% (1691 MWt) and the current rated power at 102% (1625 MWt) are negligible.

The licensee did not perform any revised analyses for the long-term containment response transients at uprated power, because of the large margins in the original analyses between the predicted and the containment design temperatures and pressures. However, the licensee did reanalyze the maximum local pool temperature at the uprated power condition and determined that the local pool temperature increased by about 1°F.

The licensee performed the peak containment pressure and temperature analysis with improved containment mass and energy release rate methodology. The staff concludes that the release rate methodology is acceptable (see NUREG-0661, containment mass and energy release). The new peak calculated pressure and temperature values are lower than the previous analysis as presented in the updated FSAR.

Based on our review, we find that the post-LOCA containment environmental conditions are not significantly affected by the proposed increase in rated power. Moreover, the licensee's latest analysis shows an increase in the margin of safety due to use of the improved calculational method.

CONSEQUENCES OF DESIGN BASIS ACCIDENT

The staff evaluation of the DAEC design basis accidents and their radiological consequences were reported in the SER dated January 1973. In that report, the consequences of the design basis accidents were calculated for the higher core power level of 1658 MWt. The staff analysis of the consequences was based on the assumption that the fuel burnup would not exceed 3800 MWd/metric ton. The licensee has assured us that the burnup of the DAEC fuel bundles will not exceed 28,500 batch average MWd/metric ton. We therefore conclude that our conclusions for accident consequences reported in our January 1973 SER remain unchanged excepting the consequences of a LOCA. The model for LOCA calculations were revised in 10 CFR 50 Appendix K subsequent to the issuance of our SER. The Appendix K requires that the consequence calculations be based on an assumed stretched power of 102% of the rated power or 1691 MWt. The staff therefore recalculated the radiological consequences of a design basis LOCA and found that the increase in the doses resulting from a LOCA will be less than 2 rems, and the resulting doses would meet all dose guidelines of 10 CFR 100.

The staff therefore concludes that the engineered safety feature designs and performances are acceptable for DAEC operation at a power level of 1658 MWt.

THE ENVIRONMENTAL IMPACTS OF OPERATION AT 1658 MWt

The operating license stage Final Environmental Statement (FES-OL) of March 1973 evaluated the potential operational impacts of DAEC at a power level of 1658 MWt. In 1981, NRC published the results of a contractor study (Reference 5) that evaluated the observed impacts of DAEC during the period 1975-1980 and compared those against the predicted impact in the construction permit stage Final Environmental Statement (FES-CP). The operation of DAEC was not found to cause any major long-term changes in aquatic resources of the site vicinity. An observed acceptable level of impact at current power, along with a predicted acceptable impact at 1658 MWt power are sufficient for us to conclude that no further review of the environmental impacts is necessary.

TECHNICAL SPECIFICATION CHANGES

1. Reactor Systems Performance

Increase in thermal power from 1593 MWt to 1658 MWt will require an increase in the nominal operating pressure/reactor dome pressure from 1005 psig to 1025 psig. This 20 psi increase in reactor operating pressure will affect reactor systems which are pressure dependent. Reactor High Pressure Scram setpoint, ATWS reactor recirculation pump trip and SRV setpoints are affected and are modified to maintain the margin above reactor operating pressure listed below:

	Uprated Power 1658 MWt (psig)	Rated Power 1593 MWt (psig)
Nominal Operating Pressure or Reactor Dome Pressure	1025	1005
Reactor High Pressure Scram Setpoint	1055	1035
ATWS Reactor Recirculation Pump Trip	1140	1120
SRV Setpoints		
Valve No. 1	1110	1080
2	1120	1090
3	1130	1100
4	1130	1100
5	1140	1110
6	1140	1110

These pressure setpoints changes are necessary for the uprated power conditions and are acceptable. The SRV pressure setpoints will improve the SRV simmer margin.

In addition, the licensee has requested that the limiting condition for operation, in the case of one ADS valve inoperable, be extended from 7 days to 30 days.

GE performed a small break LOCA analysis assuming one ADS valve inoperable. The results of the analysis meet the ECCS acceptance criteria. We find the limiting condition for operation (LCO) request acceptable.

2. Performance of the Reactor Core

The most significant reactor core related changes arising from the proposed power uprate are those to the protection system setpoints. These setpoints are adjusted to restore the operating margins to their current values or, in some cases, to increase the margins. The following core related Technical Specification changes have been evaluated:

1. Revised flow-dependent APRM scram and rod block settings, and

2. Revised Rod Block Monitor Setting

   Note that the APRM fixed trip setpoint does not require resetting since it is expressed as a percent of rated power.

The APRM trip setpoints are consistent with the analysis described above and are acceptable. The licensee has opted to use a generic cycle independent Rod Block Monitor setpoint (105 percent of rated power at rated flow), resulting in a ΔCPR of 0.19 for this event. The generic analysis is applicable for all BWR designs which use the Rod Block Monitor and is acceptable for DAEC.

3. Containment Performance

The leak rate testing for the primary containment is based upon the analysis of containment response following a DBA LOCA. As a result of the licensee's reanalysis, the peak calculated pressure is now 43 psig (whereas 54 psig was indicated in the updated FSAR). The licensee wished to incorporate the new value in the Technical Specifications. Accordingly, the licensee proposed certain changes to Section 3.7 of the Technical Specifications for the Duane Arnold Energy Center. The proposed changes are dominated by the revisions to the leak rate test pressure.

Another proposed change to the Technical Specifications involves revision of the temperature limit for conducting visual inspections of the suppression pool to be consistent with the modifications made for the Mark I improvement program. The remaining changes pertain to Section 3.7, which are clarifications or enhancements to the text.

3.0 Environmental Considerations

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

5.0 References

1. Duane Arnold Energy Center Power Uprate, General Electric Report NEDC-30603-P-1, October 1984.

2. Duane Arnold Energy Center Power Uprate, General Electric Report NEDO-30603, July 1984.

3. GESTAR II, General Electric Standard Application for Reactor Fuel, NEDO-24011-A, Latest Approved Version.

4. Approval Letter, D. G. Eisenhut (NRC) to R. Gridely (GE) dated May 12, 1978 and Supplements thereto, forming Appendix C to Reference 3.

5. NUREG/CR-2337, "Aquatic impacts from operation of three midwestern nuclear power stations. Duane Arnold Energy Center, Unit No. 1. Environmental Appraisal Report," November 1981.

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