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DUANE ARNOLD ENERGY CENTER

POWER UPRATE

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ABSTRACT

The Iowa Electric Light and Power Company (IELP) has implemented the Duane Arnold Energy Center (DAEC) Power Uprate Program with the objective of operating the DAEC at the full licensed power of 1658 MWt. Changes to the plant Technical Specifications will be necessary to attain this objective. Results of Nuclear Steam Supply System (NSSS) and containment systems safety analyses are presented. These results demonstrate that the revised Technical Specifications will satisfy the established DAEC licensing criteria, when supported by appropriate auxiliary systems analyses,and that the plant will operate at 1658 MWt without undue risk to the public health and safety. These NSSS and containment systems analyses include the following:

a. Plant heat balance at 1658 MWt and 1691 MWt
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c. Instrument setpoints at 1658 MWt
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The results of the analyses are compared to the established DAEC licensing criteria. When the accompanying Technical Specification changes are implemented, all the criteria will be satisfied.

1. INTRODUCTION

The original design basis power level of the Duane Arnold Energy Center (DAEC) was 1658 MWt, which represents an approximate 4% margin over the original commercial basis (or rated power) of 1593 MWt. This higher design basis power is reflected in the DAEC operating license which was granted at 1658 MWt and by the plant safety analyses, including reloads, which are performed at 1658 MWt, although the Plant Technical Specifications have restricted operation to the "rated power" of 1593 MWt.

This unused plant capacity and licensing basis presents an opportunity to increase the generating capacity of the Iowa Electric Light and Power Company (IELP) System with no change in calculated safety margins.

This report presents the results of analyses of the DAEC Nuclear Steam Supply System (NSSS) and containment systems necessary to demonstrate safe operation of the plant at the design basis power level of 1658 MWt. These results, when supplemented by verification analyses of plant auxiliary systems, provide justification for changing the Technical Specification maximum allowable power level to 1658 MWt.

The entire IELP program, with the ultimate objective of increasing the Technical Specification thermal power limit to 1658 MWt, is known as the DAEC Power Uprate Program. In this report, the licensed or design basis power of 1658 MWt will be referred to as the uprated power, and the current power level of 1593 MWt will be referred to as the rated power. The DAEC Technical Specifications will be revised to redefine rated power (100%) as 1658 MWt.

This increase in thermal power level will add approximately 4% to the net electrical output of the DAEC at small incremental cost and will provide significant economic benefits to the IELP rate payers.

The scope of the NSSS and containment system safety analyses, which support the increase in thermal power, was determined from reviews of the original plant licensing documentation (Reference 1), the Standard Review Plan (Reference 2), the updated FSAR (Reference 3), and selected license amendment submittals including reloads (Reference 4).

The results of the NSSS and containment systems safety analyses are compared to established DAEC licensing criteria. In all cases, the criteria are satisfied when the accompanying Technical Specification changes are implemented.

2. HEAT BALANCE

The plant heat balance defines the initial conditions and input parameters for the plant safety analyses. To determine the necessary initial conditions and input parameters, heat balance analyses of the DAEC at the uprated power level of 1658 MWt and at 102% of the uprated power level (1691 MWt) were performed. The analyses relate the reactor thermal/hydraulic parameters to the plant steam and condensate flow conditions at 1658 MWt and at 1691 MWt. The reactor heat balance parameters were used as input parameters in various analyses described in this report, including LOCA, reactor pressure vessel overpressure protection, thermal limits, reactor internal pressure differences, and containment evaluation.

A comparison of the reactor heat balance parameters at 1593 MWt (rated power condition) with those at the uprated power conditions is given in Table 2-1.

TABLE 2-1

COMPARISON OF DAEC REACTOR HEAT BALANCE PARAMETERS
FOR RATED POWER AND UPRATED POWER CASES

Parameter	Rated Power	Uprated Power	Units
Thermal Power	1593	1658	MWt
Core Flow	49.0×10^6	49.0×10^6	lb/hr
Dome Pressure	1020	1040	psia
Steam Flow	6,842,969	7,172,184	lb/hr
Core Inlet Enthalpy	526.3	528.6	Btu/lb
Feedwater Enthalpy	397.6	402.3	Btu/lb

3. POWER/FLOW MAP

The power/flow map for the DAEC at uprated power conditions is given in Figure 3-1. The map is a plot of core thermal power (in percent of uprated) versus core flow rate (in percent of rated) for various operating conditions. The power/flow map contains information on expected system performance and limits of recirculation system operation for cavitation-free operation of the recirculation pumps and the jet pumps. The shaded area indicates the region expanded by the Extended Load Line Limit Analysis (ELLLA, Subsection 12.1).

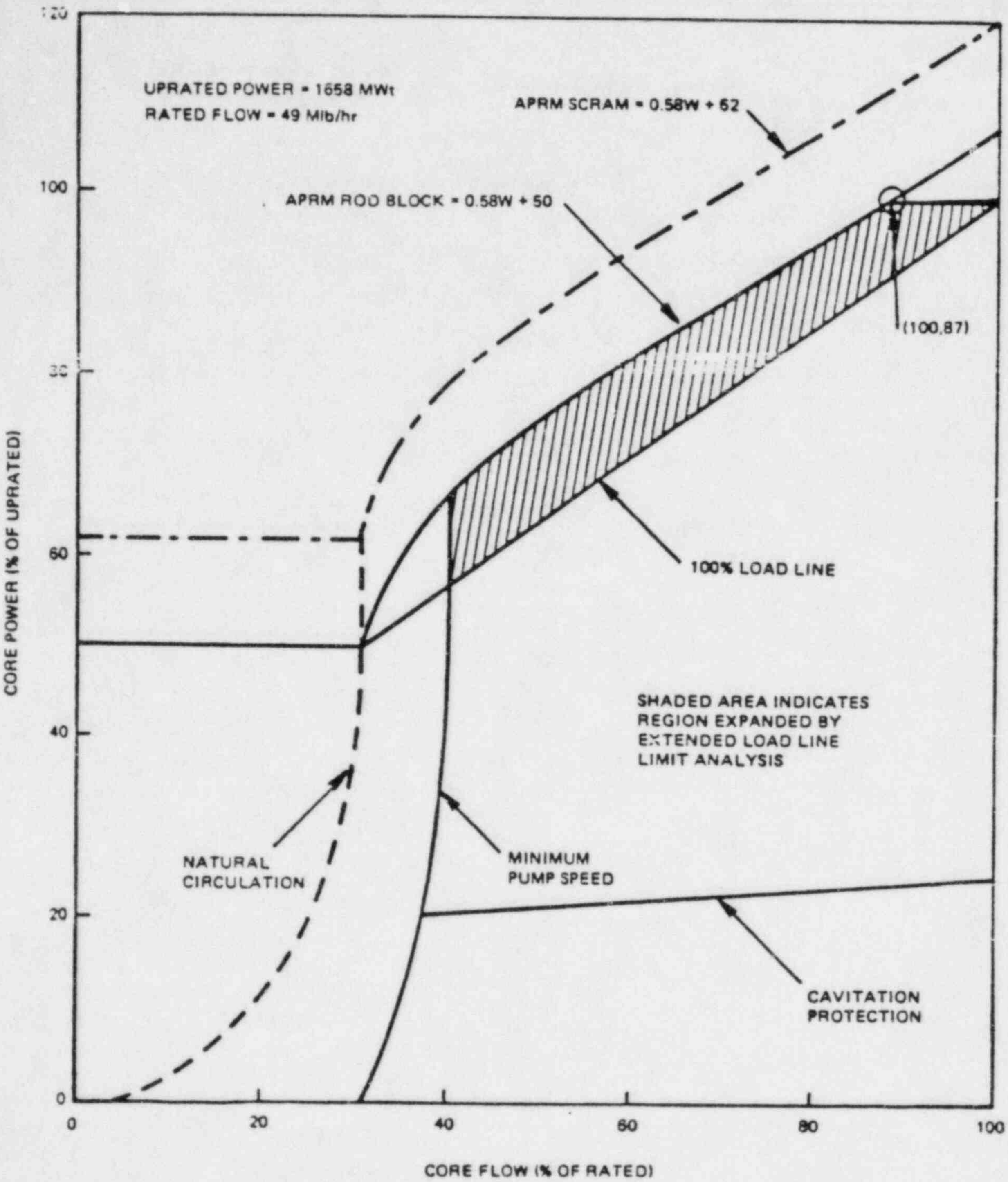

Figure 3-1. DAEC Power/Flow Map

4. INSTRUMENT SETPOINTS

Instrument setpoints are those values of sensed variables which result in initiation of protective actions and are specified in the plant Technical Specifications (Reference 5). Some instrument setpoints are dependent on the maximum thermal power and must be modified to maintain equivalent safety margins for operation at the uprated power level. Only those instrument setpoints that are affected by the reactor thermal hydraulic parameters (Section 2, "Heat Balance") will be modified; all other setpoints will remain the same.

The determination of reactor instrument setpoints is based on plant operating experience and conservative analyses. The settings are selected high enough to preclude inadvertent initiation of the protective action but low enough to assure that a significant margin is maintained between the safety systems settings and the actual safety limits.

5. LOSS-OF-COOLANT ACCIDENT ANALYSES

A Loss-of-Coolant Accident (LOCA) is a pipe break or other event resulting in an inadvertent loss of inventory and a depressurization of the reactor vessel. The analysis of the LOCA is provided to demonstrate conformance with the emergency core cooling system (ECCS) acceptance criteria of 10CFR50.46 at uprated power conditions.

The performance of the ECCS is demonstrated through application of the 10CFR50 Appendix K evaluation models and by showing conformance to the acceptance criteria of 10CFR50.46. The ECCS performance is evaluated for the entire spectrum of break sizes for postulated LOCAs. The same models were used for this evaluation as were used for previous Duane Arnold Energy Center LOCA analyses.

The applicable acceptance criteria, extracted from 10CFR50.46 are as follows:

Criterion 1: Peak Cladding Temperature
"The calculated maximum fuel element cladding temperature shall not exceed 2200°F."

Criterion 2: Maximum Cladding Oxidation
"The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation."

Criterion 3: Maximum Hydrogen Generation
"The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

Criterion 4: Coolable Geometry
"Calculated changes in core geometry shall be such that the core remains amenable to cooling."

Criterion 5: Long-Term Cooling
"After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

The results of the LOCA analyses show compliance with these acceptance criteria, and it is concluded that the ECCS will perform its function in an acceptable manner, given operation at or below the applicable maximum average planar linear heat generation rates (MAPLHGRs). Low-flow effects on LOCA analyses have been presented to the United States Nuclear Regulatory Commission (References 6 and 7) and these effects apply to the DAEC at uprated power.

6. REACTOR VESSEL OVERPRESSURE PROTECTION

The Duane Arnold Energy Center pressure relief system was designed to prevent excessive overpressurization of the primary system process barrier and the pressure vessel to preclude an uncontrolled release of fission products. The DAEC pressure relief system includes two spring safety valves (SSVs) and six dual function safety/relief valves (SRVs). These valves provide the capacity to limit nuclear system overpressurization.

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from the consequences of pressure in excess of the vessel design pressure (a more detailed discussion of the vessel pressure ASME code compliance is contained in Reference 8):

a. A peak pressure of 110% of the vessel design pressure is allowed under upset conditions (1375 psig for a vessel with a design pressure of 1250 psig).

b. The lowest qualified safety valve setpoint must be at or below vessel design pressure.

c. The highest safety valve set point must not be greater than 105% of vessel design pressure (1313 psig for a 1250-psig vessel).

Requirement "a" is conservatively evaluated by considering the most severe isolation event with indirect scram. The SRVs are assumed to be active. The SRVs open to limit the pressure rise at the bottom of the vessel to 1275 psig. There is a 100-psi margin to the vessel code limit of 1375 psig. Requirement "a" is easily satisfied and adequate overpressure protection is provided by the pressure relief system.

The two SSVs are set to actuate at 1240 psig. The proposed setpoints for the six dual function SRVs are 1110, 1120, 1130 (two valves) and 1140 (two valves) psig. These setpoints satisfy requirements "b" and "c".

7. TRANSIENTS AND ACCIDENTS

This section documents the calculated consequences of the most limiting abnormal operational transients performed for the DAEC at uprated power. The licensing basis for the transient and accident analyses for Reload 7, Cycle 8, is contained in Reference 4, and the transient and accident analyses methods are described in Reference 8.

7.1 TRANSIENTS

The scope of the transient analyses performed is identical to the reload licensing submittal (Reference 4). Requirements for vessel overpressure protection were discussed in Section 6; the evaluation of limiting abnormal operational transients from the standpoint of thermal margin requirements are discussed below. These safety requirements are based on the protection of the fuel. The most limiting abnormal operational transients for the DAEC were analyzed at the uprated conditions. The results demonstrate that the fuel cladding integrity safety limit is not exceeded.

7.2 ROD WITHDRAWAL ERROR

The rod withdrawal error (RWE) for the DAEC, Cycle 8, at uprated conditions, is based on the generic bounding analysis. The generic bounding analysis database includes a number of plants of different power densities and is valid for the DAEC at uprated power. The rod block monitor (RBM) setpoint is selected to allow for failed instruments in the worst allowable situation. It is demonstrated that even if the operator ignores all alarms during the course of this transient, the RBM will stop rod withdrawal when the critical power ratio (CPR) reaches the minimum critical power ratio (MCPR) safety limit. A more detailed discussion of analysis method is included in Reference 8.

7.3 CONTROL ROD DROP ACCIDENT

A rapid removal of a high worth control rod could result in a potentially significant excursion (i.e., insertion of reactivity). The accident which has been chosen to encompass the consequences of a reactivity excursion is the control rod drop accident (CRDA). The sequence of events and methodology is described in Reference 8. Specific analysis results for the DAEC for the uprated conditions are shown in Reference 4. The resultant peak enthalpy values are within bounding limits.

8. MCPR LIMITS

Operating limits are specified to maintain adequate margin to the onset of boiling transition during abnormal operational transients. The figure of merit utilized for plant operation is the CPR, which is defined as the ratio of the critical power (bundle power at which some point within the assembly experiences onset of boiling transition) to the operating bundle power. The critical power is determined at the same mass flux, inlet temperature, and pressure which exists at the specified reactor condition.

8.1 APPLICABILITY OF CURRENT SLMCPR AND K_F CURVES

The safety limit minimum critical power ratio (SLMCPR) and the K_f curves (MCPR multiplier) are limits that have been incorporated in the design to achieve the objective of maintaining nucleate boiling, and thus avoid boiling transition. The K_f curves, as a function of core flow, are given in Reference 8. It is concluded that the reload SLMCPR applies to uprated power as well as to the current Technical Specification power rating.

Calculations of K_f curves are normally done for steam flow conditions higher than the licensed plant power. This lends inherent conservatism to the K_f basis. The existing K_f curves are, therefore, directly applicable to uprated power operation.

8.2 MCPR OPERATING LIMIT

The DAEC uprated MCPR operating limit has been established to ensure that the fuel cladding integrity safety limit is not exceeded for any abnormal operational transient. This operating requirement is obtained by addition of the absolute maximum ΔCPR value (including any imposed adjustments factors) for the most limiting transient postulated to occur at the plant from uprated conditions, to the fuel cladding integrity safety limit.

9. STABILITY

Thermal-hydraulic stability analyses for the DAEC, Cycle 8 are performed for the uprated conditions to demonstrate analytically that no divergent oscillation or limit cycle oscillation will occur in the system.

For reload cores, two types of stability are examined utilizing a linearized analytical model. First is the hydrodynamic channel stability of one or more types of channels operating in parallel with other channels in the core. Second is the reactivity feedback stability of the entire reactor core which also involves power oscillations. The assurance that the total plant is stable, and therefore has significant design margin is demonstrated analytically when the acceptable performance limit of a decay ratio less than 1.0 is met for each type of stability.

At the most responsive condition at all attainable operating conditions, analyses have shown channel hydrodynamic and reactor conformance to the stability decay ratio criterion.

10. REACTOR INTERNAL PRESSURE DIFFERENCES/STRUCTURAL EVALUATION

10.1 REACTOR INTERNAL PRESSURE DIFFERENCES ANALYSIS

A reactor internal pressure difference (RIPD) analysis was performed for the DAEC at uprated power to confirm that the safety design bases for reactor internals are met. The analysis examined the responses of the reactor vessel internals to loads imposed during normal and accident events.

The objective of the evaluation was to determine the maximum pressure differentials across the reactor internal components during steady-state and upset conditions and during certain emergency and faulted conditions. The results of the RIPD analysis show that most pressure differences are bounded by those employed in the original component design.

10.2 STRUCTURAL EVALUATION OF DAEC AT UPRATED POWER

A structural evaluation was performed to compare the calculated peak RIPD values at uprated power to the current design values at rated power. For any of the components, where the RIPD at uprated power exceeded the current design value, the component was reanalyzed. The results of this analysis indicate that the stresses due to the new RIPD are within the allowable limits.

The impact of power uprate on the reactor vessel and the vessel nozzle design stresses were also calculated. For the reactor vessel, the vessel pressure and temperature at uprated power conditions were found to be within the values used in Reference 9. Therefore, power uprate will not violate the design stress limits for the vessel. For the vessel nozzles, only the feedwater nozzle was judged to be significantly affected by the power uprate. The results of the analysis show that the increase in feedwater flow due to power uprate will have a negligible impact on the fatigue duty of the feedwater nozzles.

The effect of increased neutron fluence on the reactor pressure vessel at uprated power was also analyzed. The DAEC operating practices, contained in References 3 and 5, were reviewed to assure the adequacy of the fracture toughness operating limits for the reactor vessel. Based on this review, it is concluded that the practices used to adjust the operating limit curve of Figure 3.6-1 of the Technical Specifications (using dosimeter measurement extrapolations to determine the fluence versus integrated power) are correct and would account for the increase in the reactor power from 1593 MWt to 1658 MWt.

11. CONTAINMENT EVALUATION

11.1 SHORT-TERM ACCIDENT RESPONSE ANALYSIS

Analyses to determine the DAEC containment short-term accident response at uprated power conditions were performed in accordance ith the load procedures of Reference 10, used in the Mark I containment DBA evaluation. In this event, an instantaneous double-ended guillotine break (DEGB) of the recirculation pump suction line is postulated to occur. Following the rupture of the recirculation line, the flow out of both sides of the break will be limited to the maximum allowed by critical flow considerations. Initial conditions of the reactor, at the time of event initiation, correspond to 102% of uprated power. Other assumptions used in the analysis are identical to those of Reference 11.

The results for the drywell and wetwell pressures and temperatures, under the uprated power design basis conditions, show that the peak drywell pressure occurs very shortly after the event initiation and is well below the containment design pressure of 56 psig. The peak drywell temperature is also well below the drywell temperature limit of 340°F.

Because the peak containment pressure for the recirculation line break is insignificantly affected by power uprate, current dynamic load analyses need not be re-evaluated. The drywell pressurization rate for the uprated power case is slightly higher than the rated power case, but it is still below the design basis limits.

11.2 LONG-TERM ACCIDENT RESPONSE ANALYSIS

The long-term containment response to transients or accidents is characterized by suppression pool temperature, containment temperature and containment pressure. Containment temperature is less than 200°F (Reference 3) compared with a design value of 281°F. Containment peak long-term pressure is less than 25 psig (Reference 3), well below the 56 psig design value. Because of the large margins between predicted and design temperatures and pressures no reanalysis was done at uprated power. Pool temperature response was evaluated for the DAEC at uprated power for the two events which result in the highest maximum local pool temperature based on previous analyses. This evaluation for uprated power at 1691 MWt (102% of uprated power) also allows for a reduction in residual heat removal (RHR) heat exchanger service water flow which results in a reduction in heat exchanger effectiveness relative to previous analyses.

The peak temperature increased by about 1°F in the new evaluation. Since the limiting local pool temperature criterion is 200.2°F, the long-term containment response shows that the DAEC conforms to the limit at uprated power.

12. IMPACT ON PLANT IMPROVEMENT PROGRAMS

12.1 EXTENDED LOAD LINE LIMIT ANALYSIS

In order to reach 100% uprated power level at high power/flow conditions without control rod withdrawals [which may be restricted by the Preconditioning Interim Operating Management Recommendations (PCIOMRs)], operation above the rated load line* is required during power ascension. This approach will minimize the capacity factor loss during startup.

The following analyses were performed as part of the Extended Load Line Limit Analysis (ELLLA) for the DAEC, Cycle 8, to verify the safe operation within the proposed extended power/flow region above the rated rod line:*

a. Thermal-Hydraulic Stability Analysis
b. Loss-of-Coolant Accident (LOCA) Analysis
c. Containment Response Analysis
d. Pressurization Transients Analysis

From the results of these analyses, it is concluded that all safety bases normally applied to the DAEC are satisfied for operation within the uprated power ELLLA region.

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* "Rated load line" or "rated rod line" refers to the power vs. flow relationship attained with a constant rod pattern that intersects 1658 MWt at rated core flow. See the DAEC Power/Flow Map of Figure 3-1.

12.2 SINGLE-LOOP OPERATION

The capability to operate using only one of the two recirculation driving loops is highly desirable in the event maintenance of a recirculation pump or other component renders one loop inoperative. In single-loop operation (SLO) the reactor would be operating at a reduced power and flow consistent with the latest recommendations (Reference 12). To evaluate the impact of uprated power on SLO, the following analyses have been reviewed for one-pump operation:

a. Loss-of-Coolant Accident
b. Stability
c. Transients and MCPR Limits

It was concluded that the results from two-loop operation analyses and previous SLO documentation justify SLO at uprated power, using the latest recommendation.

12.3 SRV LOW-LOW SET SYSTEM

The purpose of the Low-Low Set (LLS) System is to mitigate the induced thrust loads on the SRV discharge line resulting from SRV subsequent actuations during abnormal transients or small break LOCAs. For the DAEC at uprated power, two limiting events previously considered were re-analyzed.

The results of the analysis indicate that the minimum time between valve actuations of each event exceeds the minimum acceptable value by a factor of eight. It is concluded that there is no adverse effect on the LLS System due to a power uprate.

13. SUMMARY AND CONCLUSIONS

This report presents the results of analyses of the DAEC NSSS and containment systems necessary to demonstrate safe operation of the plant at the design basis power level of 1658 MWt. These results, when supported by appropriate analyses of plant auxiliary systems, thus provide justification for uprating the Technical Specification power level to 1658 MWt. A summary of the NSSS and containment systems analyses results is given in the following paragraphs.

a. Plant heat balances were performed at uprated power and 102% of uprated power. These heat balances defined steady state operating parameters and provided inputs and initial conditions for subsequent plant safety analyses. The heat balance also provides confidence that steady-state operation at uprated power can be achieved routinely.

b. The power/flow map generated provides information on expected system performance and plant operating domain at uprated power.

c. Plant instrument setpoints were examined to ensure that significant margins were maintained between the safety system settings and the actual safety limits. Certain setpoints were modified so as to maintain equivalent operating margins at the uprated power level.

d. The LOCA analyses were performed to demonstrate conformance with the Emergency Core Cooling Systems (ECCS) acceptance criteria of 10CFR50.46 for uprated power conditions. The ECCS performance was evaluated for the entire spectrum of break sizes for postulated LOCAs. Results of the LOCA calculations, at uprated power, show that the ECCS will perform its function in an acceptable manner and meet all of the 10CFR50.46 acceptance criteria, given the operation at or below the prescribed MAPLHGR limits.

e. The pressure relief system was analyzed to ensure that adequate reactor vessel overpressure protection exists during plant operation at uprated power conditions. Results of the analyses demonstrate that the ASME Boiler and Pressure Vessel code compliance criteria are met using the proposed uprated power setpoints for the SRVs and the SSVs.

f. The most limiting abnormal operational transients were evaluated for the DAEC at uprated power to ensure the integrity of the fuel cladding. Plant operation at or above the resulting MCPR limits will ensure that the fuel cladding integrity safety limit is not exceeded for any abnormal operational transient.

The CRDA was analyzed to determine the consequences of a reactivity excursion. Specific analysis results for the DAEC at uprated conditions are bounded by the rod drop design limit peak enthalpy.

g. Thermal hydraulic stability analyses were performed for uprated conditions to determine channel hydrodynamic conformance and reactor conformance to the ultimate performance criterion. At the most responsive condition, the channels and the core are within the bounds of the ultimate performance criterion of a 1.0 decay ratio.

h. A RIPD analysis was performed at uprated power to confirm that the design bases for reactor internals are met. The analysis examined the responses of the reactor vessel internals to loads imposed during steady-state and upset conditions and during certain emergency and faulted conditions. The results of the analyses show that the maximum pressure differentials across the components are within allowable stress limits.

i. Structural evaluation of the reactor internals, reactor vessel and vessel nozzles was performed at uprated power conditions. Results of this evaluation demonstrate that none of the component stresses will exceed the appropriate design limits.

j. The containment was evaluated in terms of short-term and long-term accident response at uprated power conditions. For both responses, the resulting temperatures and pressures are well below the appropriate design basis limits.

k. An ELLLA at uprated power was performed to extend the plant operating domain above the rated rod line. Stability, LOCA, containment response and pressurization transients were analyzed to verify safe operation within the extended power/flow region. From the results of these analyses, it is concluded that all safety bases normally applied to the DAEC are satisfied for operation within the uprated power ELLLA region.

l. The SLO at uprated power was analyzed to ensure safe plant performance in this mode of operation. It was concluded that the results from two-loop operation analyses and previous SLO documentation justify SLO at uprated power, using the latest recommendation.

m. The effect of power uprate on the SRV Low-Low Set (LLS) System was also analyzed. Two limiting events were considered and it was concluded that there is no impact on the LLS System due to power uprate.

These results demonstrate that the revised Technical Specifications will satisfy the established DAEC licensing criteria and, when supported by appropriate auxiliary system analyses, that the plant will operate at 1658 MWt without undue risk to the public health and safety.

14. REFERENCES

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