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DAEC Plant Support Center
Operated by Nuclear Management Company, LLC

September 20, 2001
NG-01-1123

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
NMC's Comments on the NRC Draft Safety Evaluation for the DAEC Extended Power Uprate.

References:
1. B. Mozafari (USNRC) to G. Van Middlesworth (NMC), "Duane Arnold Energy Center – Draft Safety Evaluation Regarding the Proposed Extended Power Uprate (TAC No. MB0543)," dated September 6, 2001.
2. NG-00-1900, "Technical Specification Change Request (TSCR-042): 'Extended Power Uprate'," dated November 16, 2000.

File: A-117, SPF-189

Dear Sir(s):

Per Reference 1, we were requested to review and provide comments on the Staff's draft Safety Evaluation (SE) for our Reference 2 amendment request. Attachment 1 to this letter contains a copy of the Staff's draft SE annotated with our "pen & ink" comments. Please note that Attachment 1 contains information that the General Electric Company (GE) considers to be proprietary in nature and was previously designated as such in our Reference 2 submittal and subsequent correspondence. Affidavits supporting this claim were also previously provided with that correspondence. Attachment 2 is the redacted version of Attachment 1, with the GE proprietary material removed, suitable for public disclosure.

No new commitments are being made in this letter.

Please contact this office should you require additional information regarding this matter.

Very truly yours,

Kenneth S. Putnam,
Manager Nuclear Licensing

AP01.

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Attachments:
1. NMC's Comments on NRC's Draft Safety Evaluation for the DAEC Extended Power Uprate
2. Redacted Version of NMC's Comments on NRC's Draft Safety Evaluation for the DAEC Extended Power Uprate

cc: T. Browning
G. Van Middlesworth (NMC) (w/o Attachment 2)
B. Mozafari (NRC-NRR)
J. Dyer (Region III)
D. McGhee (State of Iowa) (w/o Attachment 1)
NRC Resident Office
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Redacted Version of

NMC's Comments on

NRC's Draft Safety Evaluation for

the DAEC Extended Power Uprate

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-49

NUCLEAR MANAGEMENT COMPANY, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 OVERVIEW

1.1 Introduction

By letter dated November 16, 2000, Nuclear Management Company (NMC), the licensee, requested an amendment to Facility Operating License DPR-49 for the Duane Arnold Energy Center (DAEC). The proposed amendment would allow an increase of the authorized operating power level from 1658 megawatts thermal (MWt) to 1912 MWt at DAEC. The change represents an increase of 15.3 percent above the current rated thermal power (RTP) and is considered an extended power uprate (EPU). The amendment would approve changes to the Technical Specifications (TS) appended to the operating license to implement uprated power operation.

NMC later supplemented the original license amendment application by letters dated April 16 and 17; May 8, 10, 11, 22, and 29; June 5, 11,18, 21,and 28; July 11,19, and 25; and August 1, 10, 16, and 21, 2001.

1.2 Background

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The original rated thermal power (ORTP) for DAEC was 1593 MWt. The DAEC operating license was amended in 1985 to authorize operation at a power level of 1658 MWt. The present proposal would increase the ORTP level by 20 percent. Due to pending balance of plant modifications, the licensee proposes to implement the EPU in phases. In Phase I, the licensee plans to increase the operating power from 1658 MWt to 1790 MWt, a 112.4 percent increase from the ORTP, or a 107.9 percent increase from the current rated thermal power (CRTP). In Phase II, the licensee intends to extend the rated thermal power from 1790 MWt to 1912 MWt, a 120 percent increase from the ORTP or 115.3 percent increase from the CRTP.

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The DAEC safety analysis of the proposed EPU was provided in Attachment 6 (Reference 2) of the licensee's submittal. The attachment included General Electric (GE) Nuclear Energy Licensing Topical Report (LTR) NEDC-32980P (Proprietary), "Safety Analysis Report for Duane Arnold Energy Center Extended EPU," November 2000. Revision 1 of NEDC-32980P, dated April 2001, only changed the identification of the "stand-alone" proprietary material (i.e., the vertical "sidebars" in the margin of the text). The licensee's submittal contained plant-specific

information consistent with the scope and content of the NRC-approved GE LTR NEDC-32424P-A (Proprietary), "Generic Guidelines for General Electric Boiling Water Reactor (BWR) Extended EPU" (ELTR1), February 1999 (Reference 3), which included the staff's position paper on ELTR1 (Reference 4). For some items, the licensee referenced the analyses and evaluations in the NRC-approved GE LTR NEDC-32523P-A (Proprietary), "Generic Evaluation of General Electric Boiling Water Reactor Extended EPU (ELTR2)", February 2000 (Reference 5). [redacted] The licensee stated that the generic system and equipment performance and the generic transient and accident analyses presented in ELTR1 and ELTR2 are applicable to the DAEC EPU.

These GE LTRs provide the basis for the assessment of a licensee's request to operate at uprated power level of up to 20 percent above the plant's ORTP.

As part of the EPU review process, the staff visited the GE facility in Wilmington, North Carolina, from March 26 to 29, 2001, to audit both GE's adherence to the NRC-approved analytical methods in performing the safety analyses, and the DAEC safety analyses that support the EPU. The audit findings and their resolutions are discussed in Section 2.6 of this safety evaluation (SE).

1.3 Approach

To accomplish the EPU, the licensee proposed to implement the maximum extended load limit line analysis (MELLLA) rod line, increasing the plant's operating domain. The licensee plans to (a) generate higher steam flow though a more uniform (flattened) core power distribution, (b) increase the corresponding feedwater flow to match the higher steam flow, (c) operate the reactor along the higher MELLLA rod line and extend the MELLLA line to the EPU RTP, and (d) supply higher steam flow to the turbine generator through hardware modifications without a corresponding increase in the operating dome pressure and recirculation system flow. The licensee will achieve the EPU by revising the loading pattern of the core, by using larger batch sizes, and by introducing GE-14 fuel for EPU operation.

1.4 Evaluation of Systems, Structures, and Components

The NRC staff's review of the DAEC EPU amendment request used applicable rules, regulatory guides, Standard Review Plan (SRP) sections, and NRC staff positions on the topics being evaluated. Additionally, the staff evaluated the DAEC submittal for compliance with the generic boiling-water reactor (BWR) EPU program as defined in Reference 9. The staff also used the 1998 SE for the Monticello Nuclear Generating Plant EPU as a guide for scope and depth of review. Detailed discussions of individual review topics follow.

2.0 REACTOR CORE AND FUEL PERFORMANCE

The core thermal-hydraulic design and fuel performance characteristics are evaluated for each fuel cycle. The following sections address the effect of the EPU on fuel design performance, thermal limits, power/flow map, and reactor stability.

2.1 Fuel Design and Operation

Fuel bundles are designed to ensure that (a) the fuel bundles are not damaged during normal steady-state operation and anticipated operational occurences (AOO); (b) any damage to the fuel bundles will not prevent control rod insertion when required; (c) the number of fuel rod failures is not underestimated during accidents; and (d) the coolability of the core is always maintained. The use by each fuel vendor of NRC-approved fuel design acceptance criteria and analysis methodologies assures that the fuel bundles perform in a manner that is consistent with the objectives of Sections 4.2 and 4.3 of the SRP (Reference 13) and the applicable general design criteria (GDC) of 10 CFR Part 50, Appendix A. The fuel vendors perform thermal-mechanical, thermal-hydraulic (T/H), neutronic, and material analyses to ensure that the fuel system design can meet the fuel design limits during steady-state, AOO, or accident conditions.

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The licensee stated that the 20-percent EPU would result in a proportional increase in the average power density. However, the increase in the power density would be within the power density of an existing GE-supplied BWR. The increased operating power would affect operating flexibility and reactivity characteristics.

The licensee stated that, it will ensure that the fuel and core design limits will continue to be met at the proposed power level by varying the fuel enrichment and burnable poisons, and implementing appropriate control rod pattern management. The licensee would flatten the power distribution while limiting the absolute power in individual fuel bundles to allowable values. The licensee would use NRC-approved core design methods to analyze the core performance at the proposed EPU operation.

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[redacted] and the subsequent reload core designs at the EPU power level would take into account these limits to ensure acceptable differences between the licensing limits and the corresponding operating values.

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2.2 Thermal Limits Assessment

GDC 10 of 10 CFR Part 50, Appendix A, requires that the reactor core and the associated control and instrumentation systems be designed with appropriate margins to ensure that the specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents).

The effects of the higher MELLLA rod line and power on the thermal limits are discussed in the following sections.

2.2.1 Minimum Critical Power Ratio (MCPR) Operating Limit

The safety limit minimum critical power ratio (SLMCPR) protects 99.9 percent of the fuel rods from boiling transition during steady-state operation. The operating limit minimum critical powe ratio (OLMCPR) assures that the SLMCPR will not be exceeded as result of an AOO.

Table 9-1 of the submittal dated November 16, 2000, provides the relative SLMCPR values based on the ORTP, the cycle-specific analysis for the current RTP, and the representative equilibrium GE-14 core at the EPU power level of 1912 MWt. The table shows a slightly lower EPU SLMCPR than the SLMCPR for the current cycle. The SLMCPR is established or confirmed every reload, based on the actual core configuration and operating conditions.

The licensee also analyzed the limiting transients for operation at the EPU operating domain based on an equilibrium GE-14 core. Table 9.2 of the submittal provides the OLMCPR for the limiting transients. The licensee stated that the OLMCPR is not expected to change significantly from the values shown in Table 3-1 of ELTR1 and Figure 5-3 of ELTR2.

During the EPU audit, the staff reviewed the experimental database used for the development of the GEXL14 (critical power ratio (CPR) correlation for the GE-14 (10x10) fuel lattice design). The DAEC EPU core would introduce GE-14 fuel, and the resolution of the audit's finding ensures that the CPR correlations used to determine the MCPR are benchmarked properly. The summary of the staff's finding and GE's corrective action to resolve the findings are discussed in Section 2.X of this document.

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2.2.2 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Maximum LHGR operating limits

The MAPLHGR operating limit is based on the most limiting loss-of-coolant accident (LOCA) and ensures compliance with the emergency core cooling system (ECCS) acceptance criteria ir 10 CFR 50.46. For every new fuel type, licensees perform LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload licensees confirm that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

assuming operation

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[redacted]

in accordance with Reference 5

The licensee evaluated the plant's response to operation at the higher MELLLA rod line and power level based on an equilibrium GE-14 core. Although the Phase I reload analysis may no be based on the final EPU conditions, the Phase II reload analysis would be. The flatter power distribution may cause more fuel bundles to operate at or near the boiling transition and this could result in a slight increase in the SLMCPR. However, any SLMCPR change would constitute a TS change and the licensee would be required to submit such a change for NRC review. In addition, the audit team reviewed the GE-14 CPR correlation database used to develop the CPR correlation for the GE-14 fuel, which affects the accuracy of the SLMCPR calculations. The licensee will specify the thermal limits in the cycle-specific core operating limi reports (COLRs) as required by Section 5 of the TS. Also, the licensee cannot exceed the NRC-approved burnup limits. Therefore, the staff agrees that the licensee has appropriately considered the effects of the MELLLA/EPU operation on fuel design performance and that the thermal limits are acceptable.

2.3 Reactivity Characteristics

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2.3.1 Power/Flow Operating Map

To achieve the 20-percent increase from the ORTP, the licensee proposes to operate at the MELLLA rod line. The EPU operating domain will be defined by (a) the MELLLA upper boundary line extended up to the EPU RTP, (b) the maximum EPU power level corresponding to 120 percent of the ORTP, and (c) the existing 100 percent core flow line continued up to the EPU RTP. Therefore, the previously analyzed core flow range will be extended so that the RTP will correspond to the EPU power level and the maximum core flow will not be increased. The submittal contains the proposed EPU operating domain power/flow map.

The MELLLA upper boundary line replaces the extended load limit line analysis (ELLLA) upper boundary for SLO. The licensee stated that the maximum power statepoint for the SLO corresponding to the MELLLA upper boundary and recirculation pump speed of 102.5 percent would be to 66.8 percent of the pre-EPU RTP or 1277 MWt. The associated SLO core flow would be 53 percent core flow (25.96 Mlbm/hr). The licensee would perform the EPU SLO safety analysis based on the MELLLA state point SLO. The licensee stated that the EPU

5.1.3.4

operation at the higher rod line would also require rescaling of the associated protection system setpoints, which are discussed in Section 5.3.5 of this document.

Since the licensee is implementing the MELLLA rod line in conjunction with the EPU, the staff requested the licensee to discuss the safety analyses that were performed to demonstrate that DAEC can operate safely at the higher MELLLA rod line for offrated and rated conditions. In its August 21, 2001 response, the licensee explained that MELLLA implementation is an integral part of the EPU implementation. The DAEC PUSAR describes the details of the analyses that were performed in support of the EPU, and the licensee included in its August 21 response a list of those analyses. Based on the licensee's confirmation that the EPU safety analyses did take into account operation at the higher MELLLA rod line at the offrated conditions, the staff finds that the proposed new operating domain is acceptable.

2.4 Stability

DAEC has implemented long-term stability solution Option I-D. This Option is only applicable to plants which can demonstrate that core wide mode instability predominates and regional mode stability is not expected to occur. Small core sizes, such as DAEC, produce higher eigenvalue separation between oscillation modes and tighter core inlet orifice coefficients make regional mode oscillation unlikely. The long-term stability solutions for detect and suppress BWRs are discussed in licensing topical report, NEDO-32465, " BWR Owners' Group Stability Solutions Licensing Basis Methodology and Reload Application," published in May 1995. The Option I-D solution takes credit for unique plant characteristics which make regional mode oscillation unlikely and the stability solutions require that the existing plant instrumentation demonstrate sufficient capability for automatic detect and suppress function for core wide mode oscillation. Since the Option I-D solution is geared to respond to core wide oscillation, the flow-biased average power range monitor (APRM) flow-biased scram is used for maintaining adequate SLMCPR protection instead of the OPRM scram available in Option III. The licensee stated that for the EPU operating conditon, the flow-biased APRM flux trip provides adequate SLMCPR protection and for each fuel cycle, the MCPR safety limit protection will be demonstrated.

The Option I-D long-term solution also includes an administratively controlled exclusion region. The license uses the ODYSY code to establish the exclusion region, which is defined by a curved line that provides a constant margin to the occurrence of anticipated reactor instability. Decay ratios are calculated based on ODYSY stability criteria. The licensee stated that for Option I-D, the exclusion region boundary, those areas of the operating domain where the core decay ratio is 0.8 or greater. ODYSY calculates a best-estimate core and channel decay ratio and adds 0.15 to the core decay ratio for added conservatism. In addition, the decay ratios are calculated for state points on the power/flow map to determine the intersection of the exclusion region boundary with the natural recirculation line and the MELLLA boundary. The ODYSY stability application LTR (NEDC-32992P) has been reviewed and accepted by the staff. The review is documented in an April 20, 2001, SE. (Reference 13)

The licensee stated that the exclusion region is core and fuel dependent and it is also affected by the rated core power and the corresponding operating conditions. Therefore, the exclusion region was calculated for the EPU/fuel cycle conditions. The applicability of the current exclusion region would be evaluated for each subsequent fuel cycle.

for the vessel dome pressure scram of 1075.9 psig, plus 35 psid for the reactor bottom head, or 1110.9 psig.

2.5.2 CRD System Integrity

The licensee indicated that the control rod drive mechanisms (CRDMs) have been designed in accordance with the code of record, the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III, 1968 Edition with addenda up to and including winter 1968. The components of the CRDM form part of the reactor coolant pressure boundary (RCPB) and have been designed for a bottom head pressure of 1250 psig. This is higher than the analytical limit ~~of 1100.0 psig for the reactor bottom head pressure.~~

The licensee's evaluation indicated that the maximum calculated stress for the CRDM is less than the allowable stress limit. The analysis for cyclic operation of the CRDM resulted in a maximum cumulative usage factor (CUF) of 0.15 for the limiting CRD main flange at the EPU condition. This is less than the code-allowable CUF limit of 1.0.

On the basis of its review, the staff concurs with the licensee's conclusion that the CRDM will continue to meet its design basis and performance requirements at the EPU conditions.

2.6 DAEC EPU ONSITE REVIEW

2.6.1 Scope of Audit

During the week of March 26, 2001, four members of the NRC staff visited the Global Nuclear Fuel (GNF) engineering and manufacturing facility in Wilmington, North Carolina. The purpose of the visit was to perform an onsite review of the safety analyses and system and component performance evaluations used to support DAECs EPU. The areas covered by the review included:

1. Fuel performance information for 10X10 fuel lattice design (GE-12 and GE-14) fuel used for DAEC, including available post-irradiation examination (PIE) data;

2. Experimental database for 10x10 fuel lattice designs used for DAEC, used to develop the GEXL14 CPR correlation used for GE-14 fuel and the GEXL10 correlation for GE-12 fuel;

3. Range of experimental data and correlation fit with respect to DAEC EPU operating power, flow, and temperature requirements;

4. Statistical aspects of experimental data base and correlation (design of experiment, goodness of fit, uncertainty analysis) to support DAEC applications;

5. Design record files for pre-uprate and post-uprate DAEC LOCA analysis;

6. Review post-uprate anticipated transient without scram (ATWS) analysis for DAEC operating region;

7. Review ODYSY application to Option 1-D for DAEC application (review in conjunction with Item 9)

8. Review DAEC GE-14 fuel design compliance with respect to the approved topical reports, NEDC-32601P, NEDC-32694P, and NEDC-32502P, Rev. 1;

9. Review DAEC solution 1-D implementation for the EPU relating to:

   a) operating experience relative to thermal-hydraulic (T/H) compatibility of different DAEC fuel types at low-flow/high power conditions with off-normal void distribution,

   b) clarification of applicability of Solution 1-D to DAEC transition mixed cores, and

   c) evaluation of stability impact of changes due to DAEC mixed core with respect to restrictions in operating region and scram due to instability; and

   d) available analyses of DAEC first transition reload core design in comparison with equilibrium core, with respect to margin to operating T/H limits.

2.6.2 Findings and Resolution

The staff reviewed the experimental database used for the development of the GEXL14 CPR correlation for the GE-14 (10x10) fuel lattice design, the EPU ATWS analyses, the Option I-D long-term instability solution, and the LOCA analysis. The review resolved a number of items, but identified several issues. These issues and their resolutions are discussed below.

(1) In its CPR correlation methodology, GNF uses the computer code COBRAG to predict critical power behavior throughout the reload core fuel lattices. The staff believes that it is very difficult to predict this phenomenon in the upper portion of the core because of the very active multiple-phase transitions, and the part-length rods present in both the GE-12 and GE-14 fuels. This code has ~~never~~ not been approved by the staff for this purpose. In addition, discussions with GNF personnel indicated that COBRAG is a modified version of an original lab version of COBRA. The staff further understands that GE believes that COBRAG uses first principle models to predict boiling transition and the details of the flow field. ~~The staff's position is that this capability is currently beyond the state-of-the-art.~~ Therefore, the staff requested the licensee to provide a technical explanation as to how the COBRAG code predicts, from "first principle," the boiling transition phenomenon in the upper portion of GE-12 and GE-14 fuels.

The staff believed that GE was using data generated by the computer code COBRA G to supplement the ~~instead of~~ experimental data obtained from its CHF test facility in San Jose, CA. The use of artificial data instead of raw data raised the question of the validity of the statistical results obtained from this methodology. The statistical results are important because they are used in the calculation of the MCPR safety limit for all BWRs that use GE-14 fuel and the uncertainty associated with the data points affects the uncertainty of the safety limit calculations as well as the degree of conservatism that is used to establish the reactor operating limits. In ~~conversations with GE in late June 2001,~~ Reference 27 GE agreed to withdraw the COBRA G data from the GEXL14 data base and take appropriate action, consistent with the approved methodologies, to revise the analyses that rely on this correlation. This resolves the staff's concern.

(2) The staff questioned the adequacy of the testing of the new GE-14 fuel (and GE-12 fuel) to determine their respective CPR correlations. No power upskew or downskew experimental data was collected to develop and validate the GEXL10 or the GEXL14 correlations ~~for use in NRC Bulletin 88-07, Supplement 1, "Power Oscillation in Boiling Water Reactors (BWRs)."~~ [redacted]

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Following a GE-11 compliance audit by the staff (Team Audit of GE-11 Fuel Design Compliance with Amendment 22 of NEDE-20411-PA, 1992), GE (now GNF) was encouraged to develop a procedure for implementing Amendment 22 criteria for new correlation development as defined in GESTAR II. This was done and the procedure is documented in GNF technical design procedure TDP-0117, Rev. 2, page 8. However, the procedure does not appear to have been followed for the development of the GEXL10 and GEXL14 correlations because Items 3 and 4 were not met (there is no raw experimental data for upskew and downskew power profiles). The staff requested the licensee to provide a technical justification for not meeting these criteria of the Amendment 22 process as prescribed in the TDP. This applies to plants with GE-12 and GE-14 fuel.

which one? the GE11 or DAEC's?

In response to the NRC staff audit, GNF explained that Sections 5.3 and 5.4 of TDP-0117, Rev. 2, describe the test matrix for the ATLAS testing for the development of the GEXL correlation. This process was used, as described in "GEXL14 Correlation for GE-14 Fuel," NEDC-32851, Revision 1, September 1999. NEDC-32851, Rev. 1, also provides the process that was used to develop the uncertainties for GEXL14, using the COBRAG code to simulate the upskew and downskew power shape effects.

It appears that a piece of text is missing here?

The SLO peak cladding temperature (PCT) is first peak limited. The TLO PCT is second peak limited. There is less uncertainty in the first peak PCT calculation than in

the second peak PCT calculation [redacted] These uncertainties are reflected in the upper bound adder terms used for the first and second peak upper bound PCT calculations. [redacted] The early boiling transition occurs throughout the bundle for SLO conditions, reducing the uncertainty associated with the first peak PCT calculation for SLO. Therefore, the assumption that the upper bound adder terms used in the two-loop calculation are bounding for SLO is valid and the TLO upper bound PCT is bounding for SLO conditions. This resolves the NRC staff concerns.

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(3) The staff reviewed the design record files (DRFs) for the EPU stability and for the Cycle 18 stability calculations. Specifically, the staff evaulated whether the generic "DIVOM" curve for the core-wide mode and the regional mode oscillation specified in NEDO-32465-A (Reference 8) can be met for EPU/MELLLA operation. "DIVOM" stands for Delta critical power ratio (CPR) over Initial minimum critical power ratio (IMCPR) Versus Oscillation Magnitude (OM). The DIVOM curves are normalized curves of CPR performance versus hot-bundle oscillation. GE generated two generic curves: one for core-wide mode oscillation and one for regional mode oscillations. The curves were intended to be used in the stability licensing methodology during the reload analysis. During the audit, the staff discovered an internal GE document that questioned the applicability of the generic DIVOM curves for EPU operation using GE-14 fuel.

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On June 29, 2001, GE submitted a 10 CFR Part 21 notification regarding the use of the DIVOM curve. GE reported that stability reload licensing calculations using the generic DIVOM curve may be nonconservative for plants using the stability detect and suppress trip systems. For the Option III stability solution, the trip system setpoints, which ensure adequate MCPR safety limit protection from regional mode instability may be nonconservative. For Options II and I-D, the Part 21 report stated that flow-biased APRM flux scram may not provide adequate MCPR safety limit margin. The report stated that there is deficiency for high peak bundle power-to-flow ratios for the regional mode DIVOM curve and for high core averaged power-to-flow ratios for the core-wide DIVOM curve. GE provided a figure of merit for each generic DIVOM curve, which licensees could use to determine the applicability of the existing generic DIVOM curve for their units.

In its August 16, 2001, submittal, the licensee described its response to the Part 21 report. The figure of merit provided for Option I-D plants to determine whether the plant-specific analysis remains valid is the "core average power-to-flow ratio following a simulated flow runback on the rated rod line to approximately 30 percent of rated core flow." The figure of merit is a power/flow ratio less than 66 MWt/Mlbm/hr. For DAEC at EPU conditions, the rated rod line is essentially the MELLLA boundary as provided in the NEDC-32980P. At the bounding initial operating condition, the thermal power and core flow combine to produce a figure of merit of 64 MWt/Mlbm/hr. This satisfies the figure of merit, and therefore the EPU analysis remains valid.

DAEC has therefore demonstrated that it is not affected by the inadequacies of the DIVOM curve.

3.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

3.1 Nuclear System Pressure Relief

The safety/relief valves (SRVs) provide overpressure protection for the nuclear steam supply system (NSSS), preventing failure of the RCPB and uncontrolled release of fission products. The SRV setpoints are established to provide the overpressure protection function while ensuring that there are adequate pressure differences (simmer margin) between the reactor operating pressure and the SRV actuation setpoints. The SRV setpoints are also selected to be high enough to prevent unnecessary SRV actuations during normal plant maneuvers.

For EPU operation, the licensee will not change the SRV and safety valve (SV) setpoints because the maximum operating dome pressure will not change. The licensee evaluated the capabilities of the SRVs and SVs to provide overpressure protection based on the current setpoints and tolerances and operation at the EPU power level and determined that sufficient capability exists for overpressure protection. In addition, the licensee stated that the EPU evaluation is consistent with the generic evaluations and discussions provided in Section 5.6.8 of ELTR1 and Section 4.6 of ELTR2.

The SRV setpoint performance tests results are monitored separately to ensure compliance with the TS requirements. According to the licensee, of the 88 SRV "as found" setpoint lift verification tests that were performed from 1980 to 1999, only 7 SRVs were found to exceed their setpoints by greater than ±3 percent. Therefore, the licensee concluded that the in-service surveillance testing of the SRVs has not shown that the valves have a significant propensity for the setpoint to drift greater than 3 percent.

Table 5-1 of the EPU submittal listed the analytical limits of the SRVs and SVs, using the upper tolerance limit of 3 percent. DAEC has six SRVs and two SVs, with one SRV set to actuate at 1110 psig, a second SRV set to actuate at 1120 psig, two SRVs set to actuate at 1130 psig, and two SRVs set to actuate at 1140 psig. The two SVs are set to actuate at 1240 psig, 10 psig below the vessel design pressure of 1250 psig.

In a letter dated April 30, 1999, the licensee proposed to change the setpoint tolerance in TS Surveillance Requirement (SR) 3.4.3.1, "Safety Relief Valves (SRVS) and Safety Valves (SVs)," from ±1 percent/-3 percent "as found" to ±3 percent, "as found." The amendment request also changed TS Bases page B 3.4-19 to explain that while the SRV and SV setpoints would continue to be 3 percent for operability purposes, the valves would be set to ±1 percent during surveillance to allow for drift. In addition, the DAEC TS Section 3.4.3 limiting condition for operation (LCO) requires that the safety functions of the six SRVs and two SVs be operable. The NRC staff approved the amendment request as a plant-specific application of NRC-approved LTR NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Topical Report," which required plants to meet six conditions to justify increasing the SRV/SV setpoint tolerance. Therefore, any change in the setpoint tolerance beyond the tolerance specified in SR 3.4.3.1 would require an amendment request and

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corresponding safety analyses justifying the proposed changes. The EPU analyses are also based on the current SRV/SV setpoints and the +3 percent tolerance limit. The staff also agrees with the licensee that 7 out of 88 tests outside the tolerance band in 19 years does not demonstrate a propensity for excessive setpoint drift. Therefore, an analytical limit based on +3 percent SRV/SV setpoints at the EPU power level is acceptable to the staff.

Since the licensee performed the limiting ASME Code 1 overpressure analyses (discussed in Section 3.2 below) based on 102 percent of the EPU power level, and the current SRV/SV setpoints and upper tolerance limits will not change, the staff accepts the licensee's assessment that the SRVs will have sufficient capacity to handle the increased steam flow associated with operation at the EPU power level. The ASME Code overpressure situation is evaluated during each cycle-specific reload analysis. The capabilities of the SRVs and SVs to ensure ASME Code overpressure protection will be confirmed in the subsequent reload analyses.

3.2 ASME Code Overpressure Protection

The ASME Code allowable peak pressure for the reactor vessel and the RCPB is 1375 psig (110 percent of the design pressure of 1250 psig). This is also the acceptance limit for pressurization events. The most limiting pressurization transient is analyzed on a cycle-specific basis. This approach would be applicable for the EPU reload cycle. Section 5.5.1.4 and Appendix E of ELTR1 evaluated the ASME overpressure analysis in support of a 20 percent power increase, stating that the limiting pressurization transients events are (1) the main steam isolation valve closure (MSIVC) with failure of the valve position scram and (2) the turbine trip with bypass failure (TTNBP). The licensee analyzed both events based on an initial dome pressure of 1055 psia, 102 percent of the EPU RTP, and a representative equilibrium core. The licensee determined that MSIVC with valve position scram failure was more limiting than the TTNBP, with a 40 psid higher dome pressure. The MSIVC event resulted in a maximum reactor dome pressure of 1286 psig, which corresponds to vessel bottom head pressure of 1313 psig. Therefore, the peak calculated dome pressure (1286 psig) remains below the TS 1335 psig safety limit and the peak reactor vessel pressure (1313 psig) remains within the ASME Code limit of 1375 psig. The licensee concluded that there is no decrease in safety margin and the EPU overpressure protection analysis (given in Figure 3-1 of the submittal) is consistent with the generic analysis in Section 3.8 of ELTR2. In addition, the licensee concluded that the SRV loads for the SRV discharge line piping will remain unchanged.

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The maximum calculated pressure in the current ASME Code overpressure transient analysis meets both the ASME Code and the TS pressure limits. Therefore, the staff agrees that the licensee has demonstrated an acceptable plant response to overpressure conditions for EPU operation.

3.3 Reactor Pressure Vessel (RPV) and Internals

The NRC staff had technical issues with the methodology used to derive the fluence values used in the pressure-temperature (P-T) limits evaluation. The staff reviewed the P-T limits in a separate safety evaluation report. The fluence methodology is the subject of GE topical report NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux

Evaluations" which is currently under review by the staff. The fluence issues were discussed with the licensee in a teleconference on December 14, 2000. The staff concluded that these issues must be resolved in order to justify applying the fluence values for a full 32 effective full power years (EFPY). As an interim solution, the licensee proposed that NRC approve the P-T limits for a shorter, more defensible period. Specifically, by letter dated December 22, 2000, the license requested interim approval for the P-T curves until September 1, 2003. Further discussion of the P-T limits evaluation is contained in Section 3.3.3 of this safety evaluation.

Section Title for the above section

3.3.1 Reactor Vessel Fracture Toughness

The licensee evaluated the reactor vessel and internal components in accordance with the current design basis. The loads considered in the evaluation include reactor internal pressure difference (RIPD), LOCA, annulus pressurization (AP), jet reaction (JR), acoustic loads, therma loads, seismic loads, and fuel lift loads. The licensee indicated that the load combinations for normal, upset, and faulted conditions were considered consistent with the current design basis analysis. In the evaluation, the licensee compared the proposed EPU conditions (pressure, temperature, and flow) against those used in the design basis. For cases where the EPU conditions are bounded by the design basis analyses, no further evaluation is performed. If the EPU conditions are not bounded by the design basis, new stresses are determined by scaling up the existing design basis stresses proportionate to the proposed EPU conditions. The resulting stresses are compared against the applicable allowable values, consistent with the design basis. The staff finds that the methodology used by the licensee is consistent with the NRC-approved methodology in Appendix I of Reference 3 and is therefore acceptable.

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The stresses and CUFs for the RPV components were evaluated by the licensee in accordance with the codes of record at DAEC. These codes include the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition with addenda to and including summer 1967, 1968 Edition with addenda to and including summer 1969, and 1977 Edition with addenda to and including summer 1977, with certain exceptions and modifications as specified in the DAEC Updated Final Safety Analysis Report (UFSAR). The licensee indicated that the DAEC reactor internal components are not ASME Code components. However, ASME Code requirements have been used as guidelines in the design basis documents. The assessment is performed consistent with the current design basis. The staff finds the licensee's assessment acceptable and in compliance with the codes of record at DAEC.

move up

3.3.2 Reactor Vessel Internals and Pressure Differentials

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The licensee provided the calculated maximum stresses and CUFs for the RPV components in Table 3-2 of Reference 2. The RPV components not listed in Table 3-2 have maximum stresses and CUFs that are either not affected by the EPU or already bounded by the those listed in the table. The maximum calculated stresses shown in the table are within the allowable limits, and the CUFs are less than the code limit of unity. The maximum stresses for critical components of the reactor internals were summarized in Table 2 of Reference 6 for the EPU conditions. The calculated stresses are less than the allowable code limits shown in the table.

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The licensee indicated, in Reference 9, that the steam dryers and separators are not safety related components; however, their failure may lead to an operational concern. The licensee also indicated that, although the design basis criteria do not require evaluation of the FIV or determination of cumulative fatigue usage for the steam separators and dryers, the maximum vibration level for the shroud separators is small in comparison to the allowable limit. The licensee also indicated that the dynamic pressure loads, which may induce vibration for the dryers, are small in comparison to loads for the design basis faulted condition. ~~Accordingly, stresses in the dryers due to vibration associated with the proposed uprated condition are estimated to be less than the allowable limit.~~ In addition, the dryers will be visually inspected during their removal in each refueling outage, and any significant cracking can be detected and repaired. The design basis for the steam dryers specifies that the dryers maintain their structural integrity when subjected to a steamline break occurring beyond the main steam isolation valves. Since the dome pressure is not changed, the current steam dryer analysis remains bounding for the proposed EPU conditions. On the basis of information provided by the licensee in Reference 9, the staff concludes that the licensee has reasonably demonstrated that the steam dryers and separators will meet their design basis requirements and continue to perform their designated functions following the proposed EPU.

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Hence, the design basis loads are bounding and the dryer's structural integrity will be maintained for EPU.

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Based on its review of the licensee's evaluation of the RPV and internals, the staff finds that the maximum stresses and fatigue usage factors for safety-related components are within the code-allowable limits. The staff also concurs with the licensee's conclusion that the RPV and internal components will continue to maintain their structural integrity for the EPU condition.

3.3.3 Reactor Vessel Integrity

In Sections 3.3.1 and 3.5 of GE ~~topical~~ report NEDC-32980P, the licensee assessed the effects of the Duane Arnold EPU on the RPV and RCPB piping. With regard to the RPV, the licensee provided an assessment of (1) the impact of the EPU on the adjusted reference temperature (ART) of the limiting RPV materials; (2) the need to revise the Duane Arnold P-T limit curves; and (3) the validity of previously approved equivalent margins analyses. For the RCPB piping, the licensee provided an assessment of changes in the potential for flow-accelerated corrosion (FAC) damage due to the EPU.

For analyzing the RPV, the licensee examined the effect on the RPV fluence of operating Duane Arnold at a power of 1912 MWt until end-of-licence (EOL). The analysis addressed the expected RPV material embrittlement since it is directly related to the RPV neutron fluence, which is in turn related to the reactor operating power. The licensee stated that the estimated fluence for the EPU is conservatively increased above the UFSAR end-of-license value, and the higher fluence was used to evaluate the RPV against the requirements of 10 CFR Part 50, Appendix G. The results of the licensee's evaluation indicate that:

(a) The upper shelf energy (USE) remains bounded by the equivalent margins analysis (EMA) for the design life of the vessel and maintains the margin requirements of 10 CFR 50, Appendix G. The EMA [percent drop in] USE for the beltline materials for the limiting plate is 18 percent (less than 21 percent) and for the limiting weld is 12 percent (less than 34 percent) for 32 effective full power years [EFPY][1]

(b) The P-T curves contained in the current TS remain bounding for EPU operation up to 25 and 32 EFPY. However, the licensee has chosen to adopt ASME Code Case N-640, which requires a TS Change Request (TSCR) to modify the P-T curves. The required TSCR [was submitted by letter dated October 16, 2000].

(c) The 32 EFPY shift is increased, and consequently requires a change in the adjusted reference temperature (ART) which is the initial reference temperature of nil-ductility transition (RT_{NDT}) plus the shift [plus a margin term].

(d) The maximum operating dome pressure for EPU RTP operation is unchanged from that for current RTP operation. Therefore, the current hydrostatic and leakage test pressures are acceptable for the EPU.

The licensee concluded that the vessel remains in compliance with the regulatory requirements during EPU conditions.

The licensee's evaluation of the reactor coolant piping confirmed that changes in the flow parameter associated with the EPU would have no significant effects on the potential for FAC in those systems which might be susceptible to the phenomenon (e.g. feedwater or main steam systems).

Note: The commitment in the 12-14-00 letter was to revert to the "old" P/T curves in the Tech Specs. No limitation was made with respect to EPU operation.

As mentioned in Section 3.3, the NRC staff had technical issues with the methodology used to derive the fluence values used in the P-T limits evaluation. The fluence issues were discussed with the licensee in a teleconference on December 14, 2000. As a result of the teleconference, the license requested interim approval for the P-T curves until September 1, 2003. ~~Hence, the approval of the EPU has the same limited applicability.~~

In evaluating the effect of the EPU on the shift in ART, the staff applied the methodology found in Regulatory Guide (RG) 1.99, Revision 2, for evaluation of radiation embrittlement. The licensee submitted ART calculations and P-T limit curves valid for up to 25 and 32 EFPY. For the Duane Arnold RPV, the licensee determined that the most limiting material at the 1/4T and

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~~[1] The origin of the predicted percent drop in USE values of 34 and 21 is discussed in Section 3.0 of this SE.~~

Note: this discussion is actually on the next page.

In BWRs FAC mostly affects components made of carbon steel that are exposed to single- or two-phase flowing water at high temperature and relatively low oxygen concentration. The EPU at Duane Arnold will affect some of the plant's operating parameters, and thus change the rates of corrosion of the components subjected to FAC.

The licensee evaluated the effect of the EPU on FAC in the following systems: recirculation system, main steam and associated piping systems, feedwater, RCIC system, RVP head vent and bottom head drain, reactor water cleanup (RWCU) system, and portions of the residual heat removal (RHR) system.

The components in the recirculation system are made from stainless steel and are resistant to FAC. Therefore, they will not be affected by the EPU. However, the other systems contain carbon steel components which are exposed to single- and two-phase flowing water. The change of velocity of flow caused by the EPU will have some effect on FAC. The licensee evaluated the potential damage caused by this change, including the worst-case limiting feedwater and main steam piping flow increases. The licensee found that the change in flow is too small to have any significant effect on FAC. In addition, the fluid temperature change due to increased subcooling and reduced heat exchanger effectiveness in the RWCU system is too small to have any significant effect on FAC. The licensee also indicated that it has a program which monitors the systems where FAC is expected to occur. If any significant FAC is detected, inspection frequency will be adjusted to ensure repair or replacement of the defective components prior to reaching minimum allowable wall thickness. The staff reviewed the licensee's evaluations regarding the effects of the EPU on FAC. Based on its evaluation, the staff has concluded that the EPU has an insignificant effect on FAC in the Duane Arnold plant. In addition, any changes in FAC will be detected by the licensee's FAC program, and appropriate corrective actions will be taken.

Based on the information presented above, the NRC staff has concluded that RPV integrity, and reactor coolant system FAC issues have been adequately addressed in the Duane Arnold submittal.

3.4 Reactor Recirculation System

DAEC is currently licensed to operate up to a maximum core flow of 100 percent of the rated flow. The EPU does not require an increase in the maximum allowable core flow. Therefore, the maximum reactor recirculation flow will be maintained at the current 100 percent core flow. The primary function of the reactor recirculation system is to vary the core flow and power during normal operation. However, the recirculation system also forms part of the RCPB.

The licensee evaluated the changes in the system operating pressure and temperature at the EPU conditions and determined that changes are small and result in conditions less than the current rated conditions. The DAEC EPU will not involve any increase in the steady-state dome pressure. However, operation at the EPU RTP level would increase the two-phase flow resistance, requiring a slight increase in the recirculation system drive flow. [redacted]

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[redacted] The licensee added that the DAEC recirculation system and its components are capable of providing the core flow required for operation at the EPU RTP conditions, except for the pump motors, which are marginal in their capability. According to the licensee, the pump motor limitations are operational issues and do not affect plant safety and the recirculation system evaluations are consistent with the generic evaluation in Section 4.5 of ELTR2, Supplement 1. Section 4.5 of Supplement 1 to ELTR2 evaluated the recirculation system performance for a 20 percent EPU with a 75 psig increase in the normal dome operating pressure and concluded that the recirculation system design can accommodate the operating condition associated with the power EPU.

The recirculation pump motors may be operating slightly above their nameplate limits, but this does not directly translate to an increase in the vulnerability to trips. The PRA review of the impact of the EPU operation on initiating event frequencies and component reliability for DAEC is discussed in Section ~~3.1~~ 10.4 of this document.

From a deterministic point of view, the staff reviewed the potential impact that the recirculation pumps trip would have on the plant safety. The plant is analyzed for decreases in the reactor core coolant flow rate, which depend on the operation of the recirculation pumps and motors. The transient events in this category are (a) single and multiple recirculation pump trips, (b) recirculation flow controller failure malfunction, (c) recirculation pump shaft seizure (SLO and TLO), and (d) recirculation pump shaft break. In these events, core flow is reduced, resulting in a corresponding decrease in the reactor power. For DAEC, these transients are considered to be nonlimiting in terms of thermal limits and are not reanalyzed in cycle-specific reload analysis, ~~except for the single recirculation loop pump seizure event~~. EPU operation is not expected to make these transients limiting.

Chapter 15.3 of DAEC's UFSAR states that the pump seizure event during SLO is ~~analyzed~~ evaluated at every reload to determine the impact on the MCPR, specifically to ensure that this event does not violate the SLMCPR for the cycle. DAEC is licensed to operate with SLO, and the licensee stated that SLO operation would be limited to 66.8 percent of the EPU power level. This power level corresponds to the MELLLA upper boundary at the maximum recirculation pump speed of 102.5 percent. The submittal did not include this transient in the list of transients analyzed for the MELLLA/EPU operation. In its August 16, 2001, submittal, the licensee confirmed that the SLO recirculation pump seizure event was analyzed in support of the EPU operation.

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[redacted] The cavitation interlock, shown in the lower portion of the power/flow map, ensures that sufficient subcooling is available to prevent cavitation of the recirculation pumps. This is consistent with the evaluation in Section F.4.2.6 of ELTR1.

The licensee will not change the values (in percent) of the recirculation pump flow mismatch specification in the TS, but to be consistent with the power increase, the licensee will adjust the mismatch power basis point from 80 percent of the current rated power to 69 percent of the EPU RTP (80 percent x 1658/1912).

The DAEC TS allow the jet pump performance surveillance to be deferred for 24 hours upon reaching 25 percent RTP. The basis for the 25 percent power is for the reactor to reach stable power and flow conditions that would allow collection of meaningful data. The licensee stated that since the absolute thermal power and flow necessary for obtaining meaningful data are not changed by the EPU, the actual power level will remain the same, with 25 percent of the CRTP corresponding to 21 percent of the EPU RTP. The staff finds the licensee's assessment of the changes to the cavitation interlock, the recirculation pump mistmatch power basis, and the jet pump SR power level acceptable.

21.7

Section 4.5.3 of Supplement 1 to ELTR2 discussed the impact of 20 percent EPU on the recirculation system safety function for the (a) closure of the discharge valve during low-pressure coolant injection (LPCI), (b) pump trip in transients and ATWS, and (c) measurement of the drive flow used in the APRM flow-biased setpoint and rod blocks. For LOCA, one ~~or both~~ recirculation system discharge valve(s) must close to ensure LPCI injection into the core. Since DAEC's EPU does not involve an increase in the operating pressure, the discharge valve closure permissive pressure would not be changed or affected.

The recirculation system drive flow is measured and used as an input to the APRM for the flow-biased APRM scram and rod blocks. According to Supplement 1 to the ELTR2, the recirculation system fast transient analysis is necessary to support EPU operation for the plants that have adopted the ARTS feature to ensure adequate protection during the transient. The APRM/Rod Block Monitor TS (ARTS) program replaces the flow-biased APRM trip setdown during operation at offrated conditions. Under these conditions, ARTS plants like DAEC use power and flow-dependent MCPR, MAPLHGR and LHGR values for operation at offrated conditions. Table 9.2 of the EPU submittal provided the changes in the MCPR for the fast recirculation flow transient and indicated that the ARTS multipliers used to develop the power-dependent MCPR(P) remained bounding. This is acceptable to the staff.

3.5 Reactor Coolant Piping and Components

The licensee evaluated and the effects of the EPU condition, including higher flow rate, temperature, pressure, fluid transients ~~and vibration~~ on the piping interfaces with RCPB and the balance-of-plant (BOP) piping systems and components. The ~~components evaluated included~~ equipment nozzles, anchors, guides, penetrations, pumps, valves, flange connections, and pipe supports (including snubbers, hangers, and struts). The licensee indicated that the original codes of record as referenced in the original and existing design basis analyses, and analytical techniques were used in the evaluation. The applicable codes are the American National Standards Institute (ANSI) B31.1, ANSI B31.7, and ASME, Section III, Subsections NB and NC/ND as specified in the DAEC UFSAR. The specific codes and code editions for all piping evaluated for the EPU are provided in Reference 10 30 ~~No new assumptions were introduced that were not in the original analyses.~~ The staff finds this acceptable.

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[redacted] coolant injection (HPCI) [redacted] e licensee indicated that the evaluation follows the process and methodology defined in Appendix K of ELTR1 (Reference 3 2) and in

Section 4.8 of Supplement 1 of ELTR2 (Reference 4). In general, the licensee compared the increase in pressure, temperature, and flow rate due to the EPU against the same parameters used as input to the original design basis analyses. The comparison resulted in the bounding percentage increases in stress for affected limiting piping systems. The bounding percentage increases are compared to the design margin between calculated stresses and the ASME Code allowable limits. From the comparison, the licensee concluded that there are sufficient design margins to justify operation at the EPU condition. The bounding percentage increases were also applied to the original calculated stresses for the piping to determine the stresses at the proposed EPU condition. This approach was approved by the NRC (Reference 4) and is, therefore, acceptable to the staff.

Reference 30

~~Appendix 7 to the amendment request (Reference 1)~~ provides (a) the maximum calculated stresses and the stress ratios compared to the ASME code allowable, (b) ASME code and code edition, and (c) the CUFs (if applicable) for the piping systems evaluated for the proposed EPU. The licensee indicated that the codes and editions provided in the table are the current licensing basis codes of record. On the basis of information provided by the licensee, all calculated stresses and CUFs are within the allowable limits. The licensee also concluded that the evaluation showed compliance with all appropriate code requirements for the piping systems evaluated and that EPU will not have an adverse effect on the reactor coolant piping system design. The staff reviewed relevant portions of the evaluation provided by the licensee in Reference ~~10~~ 30 and finds the licensee's conclusion acceptable.

Note: these 3 TPs are repeated verbatim as Section 3.12 of the SE

The licensee evaluated the stress levels for BOP piping and appropriate components, connections, and supports in a manner similar to the evaluation of the RCPB piping and supports based on increases in temperature ~~and~~ pressure and flow from the design basis analysis input. The evaluated BOP systems include lines which are affected by the EPU, but not evaluated in Section 3.5 of Reference 2, such as feedwater heater piping, main steam bypass lines, and portions of the main steam, ~~recirculation,~~ feedwater, RCIC, high-pressure coolant injection (HPCI), and RHR systems outside the primary containment. The existing design analyses of the affected BOP piping systems were reviewed against the uprated power conditions. The licensee concluded that in all cases there is a sufficient margin between the calculated stresses and the code-allowable limits to accommodate the increase in stresses due to the increase in pressure, temperature, and flow as a result of the EPU. The staff finds that the stress ratios provided by the licensee are within the code-allowable limits and are, therefore, acceptable.

The licensee evaluated pipe supports such as snubbers, hangers, struts, anchorages, equipment nozzles, guides, and penetrations by evaluating the piping interface loads due to the increases in pressure, temperature, and flow for affected limiting piping systems. The licensee indicated that there is an adequate margin between the original design stresses and code limits for the supports to accommodate the load increase and therefore all evaluated pipe supports were within the code-allowable limits. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the EPU, and that no new pipe break locations were identified. The staff finds the licensee's evaluation acceptable.

The licensee indicated that the FIV levels for the safety-related MS and FW piping systems will increase in proportion to the increase in the fluid density and the square of the fluid velocity

following the proposed EPU. To ensure that the vibration level will be below the acceptable limit, the licensee is committed to perform a piping vibration startup test program, as outlined in Section 10.4.3 of the amendment submittal. The startup testing would include monitoring and evaluating the FIV during the plant startup for the proposed uprated power operation. Vibration data will be collected at interim test conditions, which correspond to 50 percent, 75 percent, and 100 percent of the ORTP, and every 5 percent step increase in power level above 100 percent of ORTP up to the final proposed EPU power level. The vibration at the new higher EPU level may be determined based on extrapolation of the vibration data taken at the lower power levels. The measured vibration levels are compared against the acceptance criteria where the allowable vibration stress levels are set by the design fatigue endurance stress intensity limits established by GE for stainless and carbon steel. The staff finds the licensee's methodology in assessing the FIV acceptable.

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3.6 Main Steam Flow Restrictors

(dome pressure)

Regarding the assessment of the main steam flow restrictor, the licensee stated that the EPU will have no impact on the structural integrity of the restrictor. In Section 3.2 of the EPU license amendment request, the licensee indicated that a higher peak RPV transient pressure of 1286 psig results from the proposed EPU conditions, but this value remains below the ASME Code limit of 1375 psig. Therefore, the main steamline flow restrictor will maintain its structural integrity following the EPU since the restrictor was designed for a differential pressure of 1375 psig, which exceeds that for the EPU condition.

Based on the above review, the staff concurs with the licensee's conclusion that the piping, components, and their supports are designed to maintain their structural and pressure boundary integrity at the proposed EPU condition.

3.7 Main Steam Isolation Valves (MSIVs)

The MSIVs are part of the RCPB and perform a safety function (steamline isolation). The MSIVs must be able to close within the specified time limits at all design and operating conditions upon receipt of a closure signal. They are designed to satisfy leakage limits set forth in the plant TS. The licensee indicated that the MSIVs have been generically evaluated, as discussed in Section 4.7 of ELTR2 (Staff Safety Evaluation of GE Licensing Topical Report NEDC-32523-P, "Generic Evaluations of General Electric Boiling Water Reactor Extended EPU," including Supplement 1, Volumes I and II). This evaluation covers both the effects of the changes to the structural capability of the MSIV to meet pressure boundary requirements and the potential effects of EPU-related changes to the safety functions of the MSIVs. The licensee stated that the conditions for DAEC are bounded by those in the generic analysis. Although the dome pressure does not increase with this EPU, the increase in flow rate assists MSIVC, which results in a slightly faster MSIVC time. The actual in-plant settings will be adjusted such that the closure speed will be maintained within the required range. Therefore, the MSIVs remain acceptable for EPU operation.

Based on our review of the licensee's rationale and evaluation, we concur with the licensee's conclusion that EPU operation as indicated above remains bounded by the conclusion of the generic evaluation in Section 4.7 of ELTR2 dated September 14, 1998, and that the plant

Note: This is the date on the letter for the SER to ELTR-2, Supp. 1 (Ref. 4)

operations at the proposed EPU level will not affect the ability of the MSIVs to perform their isolation function.

3.8 RCIC System

Note: Per telecon with the NRC Staff on 9-19-01, portions of this section will be re-worded.

The RCIC provides core cooling when the RPV is isolated from the main condenser and the RPV pressure is greater than the maximum allowable for starting the low-pressure core cooling systems. [redacted]

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Section 3 of ELTR2 provided an evaluation of the RCIC design basis and provided a generic LOFW analysis for all BWR types. The report states that for BWR 4, 5, and 6 plants, the RCIC system (the smaller of the two high-pressure supply systems) should maintain the reactor water level so that the very low level trip setpoint (Level 1) for the low-pressure ECCS, would not trip, and the MSIVs would not close. The document reports that the operational aspects will be evaluated for the Level 1 setpoint for each plant and documented in the plant-specific licensing report. [redacted] For the LOFW transient without immediate isolation, pressure control would be achieved through the main steam bypass system to the main turbine condenser and the RCIC or HPCI would provide sufficient water to maintain core coverage. Since this is an operational

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issue and would not affect RCIC's ability to provide core cooling and maintain the water level above the TAF, the staff accepts the licensee's assessment.

Section 4.2.3 of Supplement 1 to ELTR2 discusses the potential for turbine overspeed of the steam-driven RCIC and HPCI turbines and pumps. The supplement stated that startup transients for the HPCI and the RCIC systems at a potentially higher steam inlet pressure may result in an increased initial turbine acceleration rate, increasing the peak initial speed and the probability of the system trip. The HPCI and RCIC startup transients are dependent on the reactor pressure and the supplement recommended that the modifications described in SIL 377, "RCIC Startup Transient Improvement with Steam Bypass," be implemented to assure RCIC and HPCI availability for EPUs that involve a reactor pressure increase. [redacted]

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The licensee evaluated the NPSH available for the RCIC and stated that the EPU does not decrease the NPSH available for the RCIC pump or increase the NPSH required, beyond the specified design basis value. For events such as fires and ATWS, if the suppression pool temperature is greater than the operational limit, the RCIC system can use the dedicated condensate storage tank volume as source of water.

The licensee determined that the RCIC system has been evaluated for the LOFW transient event and the evaluation is consistent with the conclusions of Section 4.2 of ELTR2. [redacted] The staff concurs with the licensee's evaluation.

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3.9 Residual Heat Removal System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel, and to remove sensible and decay heat from the primary system and containment following reactor shutdown for both normal and post-accident conditions. The RHR system operates in the LPCI mode; shutdown cooling mode; suppression pool cooling mode; containment spray cooling mode; and fuel pool cooling assist mode.

3.9.1 RHR System - Suppression Pool Cooling Mode

The ultimate heat sink for DAEC is the Cedar River. During plant operation, heat loads from both safety-related and non-safety-related systems are rejected to plant cooling towers (normal heat sinks). Following a LOCA, the ESW and RHRSW heat loads are preferrably to be rejected back to the plant cooling towers. However, if the cooling towers are not in operation or not operable, the total ESW and RHRSW heat loads are rejected to the UHS. The licensee performed an evaluation and concluded that the UHS will provide a sufficient quantity of water at a temperature of 95 °F (design temperature) following a design basis accident. The total heat rejected by the ESW system and the RHRSW system has a negligible thermal effect at the discharge structure.

Based on our review of the licensee's evaluation and rationale, and the experience gained from our review of EPU applications for other BWR plants, the staff find that DAEC operations at the proposed EPU will have an insignificant impact on the UHS.

6.5 Standby Liquid Control (SLC) System

The licensee evaluated the effect of the EPU on the SLC system injection and shutdown capability. The DAEC SLC system is a manually operated system that pumps sodium pentaborate solution into the vessel in order to provide neutron absorption and bring the reactor to a subcritical shutdown condition from RTP.

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[redacted] According to the licensee, a new fuel design combined with an extension in the fuel cycle operating time would require an increase in the minimum reactor boron concentration from 600 ppm to 660 ppm. By increasing the volume of the stored boron solution, the licensee can provide the required amount of boron for the EPU cycle. The licensee has submitted, and the staff has approved, an amendment requesting an increase in the SLC solution volume separate from the EPU submittal.

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[redacted]

The SLC ATWS performance is addressed in Section 9.3.1 and the licensee has stated that the evaluation is based on a representative core design at the EPU conditions. The licensee determined that the ATWS analysis showed that there is no adverse effect on the ability of the SLC system to mitigate an ATWS. Therefore, the capability of the SLC to perform its function is not affected by the EPU.

The staff asked the licensee to confirm that for all limiting ATWS analyses, the SLC system would be able to inject the required flow rate at the required time in the analyses without lifting

of the SLC system relief valve. In its August 16, 2001, submittal, the licensee responded with an explanation of the sequence of events for the bounding [redacted]

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Considering that the ODYN calculation is very conservative (plant response to the water level transient is expected to be considerably faster than ODYN models it to be), that there will still be [redacted] the staff agrees with the licensee's conclusion that the SLC system will be able to inject boron into the RCS as required by 10 CFR 50.62.

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6.6 Power-Dependent HVAC Systems

The HVAC systems consists mainly of heating, cooling supply, exhaust, and recirculation units in the turbine building, reactor building, and drywell. Plant operation at EPU conditions is expected to result in slightly higher process temperatures and higher electrical currents in some motors and cables. The licensee stated that HVAC systems affected by the EPU are those systems in the turbine building, reactor building, main steam tunnel, and drywell.

In the turbine building, the areas affected by the EPU include the feedwater heater bay and condenser areas. Other areas are unaffected by the EPU because the process temperatures remain relatively constant. Due to EPU, the temperature at the feedwater heaters in the turbine building increases approximately 5 ~~percent~~ °F relative to the current operating temperature.

Heat loads in the drywell increase slightly due to increase in the recirculation pump motor horsepower and the feedwater process temperature. The maximum temperature increase is 1.3 °F. The increases in the condensate pump motor, feedwater pump motor, and reactor recirculation motor generation (MG) set motor horsepower result in a temperature increase that is less than 2 °F.

The heat loads discussed above represent an increase of 2 percent to 5 percent in the drywell, reactor building, and main steam tunnel, and approximately 21 percent in the heater bay area. The licensee stated that based on the review of design basis calculations and environmental qualification design temperatures, these heat load increases are within the excess design margin. By letter dated July 27, 2001, in response to the staff's RAI, the licensee stated that the numerical values for increases in normal containment temperature are derived from Duane Arnold plant-specific calculations. These plant-specific calculations are documented in General Electric's design record files.

25 (Reference 28)

Note: No discussion of LPCI or Shutdown Cooling modes of RHR?

The suppression pool cooling (SPC) mode of the RHR system is designed to remove heat discharged into the suppression pool to maintain pool temperature below the TS limit during normal plant operation and below the suppression pool design temperature limit after an accident. The EPU increases the reactor decay heat, which increases the heat input to the suppression pool during a LOCA, which results in a higher peak suppression pool temperature. The EPU effect on suppression pool cooling after a design basis LOCA is addressed in Section 4.1.1.

3.9.2 RHR System - Containment Spray Cooling Mode

The containment spray cooling mode (CSC) mode is designed to spray water from the suppression pool via spray headers into the containment airspace, to reduce the long-term containment pressure and temperature during post-accident conditions. The EPU slightly increases the containment spray water temperature. This increase has a negligible effect on the use of the CSC mode to maintain the containment pressure and temperature within their design limits as the peak pressure and temperatures are reached well before the use of the RHR system in the CSC mode is assumed to occur.

Based on our review of the licensee's rationale and evaluation, we concur with the licensee that plant operation at the proposed EPU level will have an insignificant impact on the CSC mode.

3.9.3 RHR System - Fuel Pool Cooling Assist Mode

For plant operation at the proposed EPU, the decay heat load for any specific fuel discharge scenario will increase. In the event that the spent fuel pool (SFP) heat load exceeds the heat removal capability of the SFP cooling system (i.e. during full-core offload events), the RHR system will be operated in the fuel pool cooling assist mode to provide supplemental cooling to the SFP, and to maintain the SFP temperature within acceptable limits. Section 6.3 addresses the adequacy of the combined heat removal capability of the SFP cooling system and the RHR system operating in the fuel pool cooling assist mode to meet the increases in SFP heat loads resulting from the proposed EPU.

3.10 Reactor Water Cleanup System

Note: This refers to only the piping stress evaluation. No mention of the system functional design basis for water chemistry?

Evaluation of the reactor water cleanup system (RWCU) is included in Sections 3.3.3 and 3.5 of this SE.

3.11 Main Steam and Feedwater Piping

The main steam and feedwater piping evaluation is addressed along with reactor coolant piping in Sections 3.3.3 and 3.5 of this SE.

3.12 Balance-of-Plant Piping

3.12.1 Pipe Stress Evaluation

The licensee evaluated the stress levels for BOP piping and appropriate components, connections, and supports in a manner similar to the evaluation of the RCPB piping and supports based on increases in temperature ~~and~~ pressure and flow from the design basis analysis input. The evaluated BOP systems include lines which are affected by the EPU, but not evaluated in Section 3.5 of Reference 2, such as feedwater heater piping, main steam bypass lines, and

code, the M3CPT code. These analyses will cover the response through the time of peak drywell pressure throughout the range of power/flow operating conditions with EPU.

Appendix G of NEDC-32424P-A also requires the applicant to perform long-term containment heatup (suppression pool temperature) analyses for the limiting UFSAR events to show that pool temperatures will remain within limits for suppression pool design temperature, ECCS NPSH, and equipment qualification temperatures. These analyses can be performed using the GE computer code SHEX. The SHEX computer code has been used by GE on all BWR EPUs.

The staff requested the assistance of Information Systems Laboratories, Inc. (ISL), in performing audit calculations of both the short-term and the long-term responses of the Duane Arnold containment to a double-ended guillotine break of a recirculation line. These calculations used mass and energy input values provided by the licensee and plant description values furnished by the licensee and obtained from the DAEC UFSAR. The staff used conservative assumptions in performing these calculations. The results of the staff calculations for both the short term (peak pressure and drywell temperature) calculations and the long term (peak suppression pool temperature) calculations agree well with the licensee's results for the trend and timing of important parameters. The numerical values of two calculations are close and can be explained by small changes in any one of several input values. ISL's report containing the details of the calculations is available in ADAMS with accession number.....

4.1.1 Containment Pressure and Temperature Response

Short-term and long-term containment analyses results following a large break inside the drywell are documented in the DAEC UFSAR. The short-term analysis was performed to determine the peak drywell and wetwell pressure response during the initial blowdown of the reactor vessel inventory into the containment following a large break inside the drywell (DBA-LOCA), while the long-term analysis was performed to determine the peak pool temperature response considering decay heat addition.

The licensee indicated that the containment analyses were performed in accordance with NRC guidelines using GE codes and models. The M3CPT code was used to model the short-term containment pressure and temperature response. The licensee also indicated that the SHEX code was used to model the long-term containment pressure and temperature response for EPU.

Note: The PUSAR also discusses a "simplified" calculation that was done for the very long-term results.

4.1.1.1 Long-Term Suppression Pool Temperature Response

(a) Bulk Pool Temperature

The licensee indicated that the long-term bulk suppression pool temperature response with the EPU was evaluated for the DBA-LOCA. The bounding analysis was performed at 102 percent of EPU RTP and assumes that no offsite power is available following the LOCA with only minimum diesel power available. This results in only one RHR loop with one RHR pump, its heat exchanger, two RHR service water pumps , and one core spray pump being available. No credit is taken for use of containment spray. The analysis was performed using the SHEX code and the more realistic decay heat model (ANS/ANSI 5.1-1979 + two sigma) with a core average exposure of 31.7 GWD/Short Ton (corresponding to a core average time at power of 3.5 years),

subsequent actuations has been reduced to its initial condition. The EPU has an insignificant impact on the time intervals between SRV openings, and therefore, it has an insignificant impact on the loads from subsequent SRV actuations.

Based on our review of the licensee's rationale and evaluation, the staff agrees with the licensee's conclusion that with the low-low set logic implementation, the EPU will have insignificant or no impact on the SRV containment loads.

4.1.2.3 Subcompartment Pressurization

The licensee indicated that the calculations to determine the asymmetrical loads on the vessel, attached piping, and biological shield wall due to postulated pipe break in the annulus between the reactor vessel and biological shield wall have been updated to treat the break flow as subcooled liquid. This conservative change in methodology reduces the design margin more than the effect of the EPU. It was also found that the original analysis had not been updated during the previous EPU. For the limiting case of the recirculation outlet nozzle shield plug, the pressurization transient inside the shield wall increases peak pressure from 16.5 psi in the original analysis to 18.98 psi including all the above changes and remains below the design pressure of 20 psi. The results of the updated calculations including the effects of the EPU indicate that the biological shield wall and component designs remain adequate, because there is sufficient pressure margin available.

power uprate

16.25 psi

Based on the above evaluation, the staff concurs with the licensee's conclusion that the subcompartment pressurization will remain acceptable after the EPU.

4.1.3 Containment Isolation

The licensee indicated that the system designs for containment isolation are not affected by the EPU. The capability of the actuation devices to perform with the higher flow and temperature during normal operations and under post-accident conditions has been determined to be acceptable.

Based on our review of the licensee's rationale and evaluation, and the experience gained from our review of the EPU applications for other BWRs, the staff agrees with the licensee's conclusion that the containment isolation system will remain acceptable for plant operations at EPU.

4.1.4 Generic Letter 89-16

The licensee indicated that in response to Generic Letter (GL) 89-16, DAEC installed a hardened wetwell vent system. The criteron specified in GL 89-16 for the hardened wetwell vent was the ability to exhaust energy equivalent to 1 percent RTP at a pressure of 56 psig. The design of the hardened wetwell vent was based on the current power level of 1658 MWT. Based on the as-built design, the hardened wetwell vent will exhaust approximately 0.91 percent of the proposed rated thermal power of 1912 MWt.

The licensee indicated that the primary objective of the hardened wetwell vent is to preclude primary containment failure due to overpressurization, given a loss of decay heat removal

event. The staff considers the new value of 0.91 percent RTP to be a change to the design basis. However, this change is acceptable since there is still adequate margin between the 0.91 percent RTP criterion and the value of decay heat at the time the licensee would initiate venting.

Based on our review of the licensee's rationale and evaluation, the staff agrees with the licensee's conclusion that the hardened wetwell vent system will remain acceptable for preventing containment overpressure for plant operation at EPU with the revised design basis value of 0.91 percent RTP.

4.2 Emergency Core Cooling System

50.67

The ECCS is designed to provide protection in the event of a LOCA due to a rupture of the primary system piping. Plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on: (a) the peak cladding temperature, (b) local cladding oxidation, (c) total hydrogen generation, (d) coolable core geometry, and (e) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system piping. Assuming a single failure of the ECCS, the LOCA analyses identify the break sizes that most severely challenge the ECCS systems and the primary containment. The MAPLHGR operating limit is based on the most limiting LOCA analysis, and licensees perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS for DAEC includes the HPCI, the LPCI mode of the RHR, the low-pressure core spray system and the ADS. The ADS system is discussed in Section 4.2.4, and the ECCS performance in Section 4.3.

4.2.1 High Pressure Coolant Injection System.

The HPCI system (with other ECCS as backup) is designed to maintain reactor water level inventory during small and intermediate break LOCA, isolation transients- and LOFW. For a large break LOCA, the reactor will depressurize rapidly, thereby rendering the HPCI system inoperable.

The licensee stated that the HPCI system is designed to inject with a maximum reactor pressure of 1120 psig and this is consistent with the current DAEC licensing basis, which allows for an increase of the as-found SV and SRV safety mode setpoint from +/- 1 percent to +/-3 percent. The HPCI system is required to start and operate reliably over its design operating range. The licensee stated that since there are no increases in the SRV setpoint pressures with the EPU, no changes are required to the HPCI system startup transient or system reliability from the currently licensed pre-EPU conditions. During the LOFW event, and isolation transients, the HPCI and/or RCIC maintain water level above the TAF. The SRVs would open and close as required to control pressure and HPCI or RCIC will eventually restore water level.

Previously, the licensee has implemented the turbine control modifications recommended by GE services information letter (SIL) 480. According to the licensee, these modifications

minimize the impact of reactor pressure increases on: (a) the magnitude of the turbine peak initial speed, (b) the initial pump discharge pressure, and (c) the pump discharge flow rate. The modifications described in the SIL hydraulically limit the opening of the control valves during the initial startup period, thereby minimizing the potential for turbine overspeed. Since the HPCI turbine has been modified as recommended in GE SIL 480, increasing the availability of the HPCI system during isolation transients including LOFW, the staff accepts the licensee's assessment regarding HPCI turbine overspeed.

The licensee evaluated the NPSH available for the HPCI system for EPU operation and stated that the EPU does not decrease the NPSH available for the HPCI pump or the NPSH required, beyond the specified design basis value. For events such as fires and ATWS, if the suppression pool temperature is greater than the operational limit, the licensee stated that the HPCI system can use dedicated condensate storage tank as a source of water.

The licensee evaluated the capability of the HPCI system, for operation at the EPU power level, to provide core cooling to the reactor: (a) to prevent excessive fuel PCT following small and intermediate break LOCA, and (2) to ensure core coverage up to the TAF in isolation transients and LOFW transients. The licensee stated that the HPCI evaluation is applicable to and consistent with the evaluation in Section 4.2 of ELTR2, except for the DAEC HPCI system design basis maximum system pressure. The licensee determined that the HPCI system is acceptable for the EPU.

lower

The generic evaluation in Section 4.2 of the supplement to ELTR2 is based on typical HPCI and RCIC pump design pressures of 1500 psig and pump discharge piping pressures between 1500 psig and 1280 psig. The DAEC maximum reactor pressure for design basis injection for both high pressure systems is lower than typical BWRs. However, the licensee evaluated the capability of the HPCI system to perform as designed and analyzed its performance at the EPU conditions, and concluded that HPCI system can start and inject the required amount of coolant into the reactor for the range of reactor pressures associated with LOCA and isolation transients. The staff agrees with the licensee's assessment.

This is not a true statement. Note: the comparison with the ELTR statement is not valid; the pressure rating of the piping is not the same as the maximum pump discharge pressure.

4.2.2 Low Pressure Coolant Injection

The LPCI mode of the RHR is automatically initiated in the event of a LOCA, and in conjunction with other ECCS system, the LPCI mode is required to provide core cooling for ECCS events. The licensee stated that, as indicated in the ECCS performance discussion in Section 4.3, the calculated LOCA PCT would increase slightly for the EPU. However, the existing LPCI system combined with other ECCS systems provide adequate long term post-LOCA core cooling. The licensee added that the existing RHR hardware has the capability to perform the design injection function of the LPCI mode for operation at the EPU condition and the generic evaluation in Section 4.1 of ELTR2 bounds the DAEC LPCI system performance. The staff finds the evaluation acceptable.

4.2.3 Core Spray System

The core spray system initiates automatically in the event of a LOCA and in conjunction with other ECCS systems, the core spray system provides core cooling for all ECCS events.

The licensee stated that, as indicated in the ECCS performance discussion in Section 4.3, the calculated LOCA PCT would increase slightly at the EPU. However, the existing core spray system, combined with other ECCS systems, will provide adequate long term post LOCA core cooling. The licensee added that the existing core spray system hardware has the capability to perform its design injection function at the EPU conditions and that the generic evaluation in Section 4.1 of ELTR2 bounds the DAEC core spray system performance. The staff finds this acceptable.

4.2.4 Automatic Depressurization System (ADS)

The ADS uses the SRVs to reduce reactor pressure after a small-break LOCA with HPCI failure, allowing LPCI and core spray to provide cooling flow to the vessel. The plant design requires the SRVs to have a minimum flow capacity. After a delay, the ADS actuates ~~either on low water level plus high drywell pressure or~~ on low water level ~~alone~~. The licensee stated that the ability of the ADS to initiate on appropriate signals is not affected by the EPU. However, the EPU decay heat is higher, increasing the required flow capacity. The licensee stated that the increase in the required flow capacity is within the current system design capability. The staff accepts the licensee's evaluation.

4.2.5 Net Positive Suction Head

The licensee indicated that the available and required NPSH for the HPCI pump are not changed for the EPU. The available NPSH is based on the TS limit of 120 °F for the suppression pool temperature that requires depressurization of the RPV. This TS limit is not changed and there are no physical changes being made in the HPCI system.

The licensee also indicated that the NPSH requirements for the RHR and core spray ECCS pumps are calculated from the EPU long-term suppression pool temperature response. The most limiting case for NPSH occurs at the peak long-term suppression pool temperature of 209.2 °F and the wetwell pressure of 28.0 psia (13.3 psig) with one RHR pump and one core spray pump operating at rated flows. The result of this analysis establish that a containment pressure of 20 psia is adequate to ensure NPSH is available to the low pressure ECCS pumps post-LOCA during EPU operation.

During the initial review of the DAEC operating license, the staff concluded that a limited amount of containment overpressure was required to ensure adequate NPSH for one core spray and one RHR pump during the long-term transient following the design basis LOCA. In the licensing safety evaluation report, the staff noted that the design did not meet the guidelines of RG 1 but a containment pressure margin of 2.7 psi would exist to ensure adequate NPSH. This margin was depicted on UFSAR Figure 5.4-15. In the current licensing basis, containment pressure required (i.e., containment overpressure) to maintain minimum NPSH for the core spray pumps is 17.8 psia (25.3 psia available) at the peak suppression pool temperature of 202.7 °F. Thus, implementation of the EPU will require an increase of 2.2 psi in containment overpressure for NPSH over the current conditions. According to the licensee, there is sufficient containment pressure available for post-LOCA conditions to establish the margin between the 20.0 psia required and the 28.0 available to satisfy the 2.7 psid margin required by the current licensing basis. Therefore, the licensee concluded that adequate NPSH is ensured under EPU conditions. The licensee also indicated that the maximum containment

overpressure for NPSH will be required after approximately 8 hours after a DBA-LOCA. Reliance on the containment overpressure decreases as the temperature of the suppression pool decreases, if all other factors are held constant.

Note: The requested curves were provided by the DAEC for the purposes of preparing for the ACRS Thermal-hydraulics subcommittee meeting.

The staff requested that DAEC provide NPSH curves which depict the amount of time the peak containment overpressure must be relied upon. This information is important to demonstrate that the risk associated with the limited time the containment overpressure is required is low relative to the potential loss of containment integrity during the same time period. The licensee disagrees that such curves are required. The licensee's position is that an overpressure margin is an acceptable criterion for adequate available NPSH. This means that as long as the margin is maintained, the absolute amount of overpressure can be increased. The staff disagrees with this position and considers it to be an open item.

4.3 ECCS Performance Evaluation

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

Note: All currently-used fuel types (GE10, GE12 & GE14) were analyzed. Only the most-limiting (GE14) was presented in the PUSAR.

The licensee performed the LOCA analysis at 102 percent of the EPU RTP, using GE-14 fuel. The ECCS-LOCA analysis was based on an NRC-approved methodology (SAFER/GESTR) and the submittal summarized the results in Table 4-3. The licensee determined the licensing basis PCT at the rated core operating conditions with an adder to account for the uncertainties. For the EPU conditions, the licensing basis PCT based on the limiting GE-14 fuel design is less than 1510 °F at rated flow in comparison with the pre-EPU PCT of 1500 °F . The estimated upper bound PCT for the limiting GE-14 fuel is less than 1350 °F for the EPU conditions, which is below the 1600 °F limit required by the NRC SE for the SAFER/GESTR methodology.

For SLO, the licensee applied a multiplier to the TLO MAPLHGR limits. The licensee stated that the multiplier to the MAPLHGR for the SLO operation ensures that the SLO nominal PCT is less than the PCT for the nominal TLO. Section 2.6 discusses the findings from the staff audit of these calculations and the licensee's response.

Table 4-3 of the submittal indicates that long-term decay heat removal requirements can be satisfied for the EPU conditions either by having the core reflooded above the TAF or by having the core reflooded to the top of the jet pumps with one core spray pump in service. The licensee determined that the ECCS performance under LOCA conditions, and the analysis models, satisfy the requirements of 10 CFR 50.46 and Appendix K.

In addition, as part of the EPU review process, the NRC staff audited the DAEC LOCA analysis. The staff focused on the GNF's use of the LOCA codes and their applicability to Duane Arnold. The staff examined the design record files describing the LOCA, which contained both the pre-EPU and the EPU analyses, and made the following observations:

1. The analyses were based on the NRC approved SAFER/GESTR methodology and GNF followed NRC approved process in performing the ECCS-LOCA analysis.

24 hours prior to the 2.3 day mark when the CAD system would be required to inject nitrogen to maintain containment below the 5 percent oxygen limit.

By letter dated July 25, 2001, in response to a staff's request, the licensee indicated that this condition is a deviation from its previous commitments to both NUREG-O737, Item II.F.1(6), and RG 1.97 regarding continuous hydrogen monitoring. During the initial 22-hour period, the accuracy of these monitors will not conform to the NUREG-0737 requirement, but the monitors will be available. The stated temperature effect causes the monitors to read in the conservative direction, i.e., higher than actual hydrogen and oxygen concentration. Thus, to preclude action to actuate the CAD system prematurely, it is chosen administratively not to take direct actions based solely on these hydrogen and oxygen monitors during the first 24 hours. While the monitors will not be used to obtain an absolute reading during this initial accident period, they will provide valuable trending information if hydrogen and/or oxygen concentration changes are occurring inside the containment. As per DBA analysis, a combustible gas mixture will not be generated for approximately 2.3 days. Consequently, this potential inaccuracy in these monitors during the first 22 hours will not hinder the ability to effectively execute the emergency plan. The emergency response facilities have sufficient information available, using the Severe Accident Management Guidelines, to diagnose whether a flammable mixture is developing inside the containment and recommend that the control room take appropriate actions to protect the containment. Based on the above, we find the proposed deviation for the initial 22 hours after a DBA from the previous commitment to NUREG-0737, Item II.F.1(6), and RG 1.97 regarding accuracy of continuous hydrogen monitoring is acceptable as stated.

Note: We asked for a deviation for 24 hours.

Based on our review of the licensee's rationale and evaluation, the staff concludes that the operation of the post-LOCA combustible gas control system will remain acceptable at the EPU conditions.

4.5.3. Emergency Cooling Water System

The staff's evaluation of the emergency cooling water system is addressed in Section 6.4 of this SE.

4.5.4 Emergency Core Cooling Auxiliary System

Not applicable to DAEC.

4.5.5 Control Room and Technical Support Center Habitaility

The control building ventilation system (CBVS) processes the control building intake atmosphere to limit the release of radioisotopes to the control room that may leak from containment under DBA-LOCA conditions. The capacity of the CBVS provides a positive differential pressure between the control room and the outside environment to minimize the potential for unprocessed in-leakage into the control room. The licensee stated that this capability is not affected by the EPU.

The technical support center (TSC) air cleanup system (ACS) processes the TSC intake atmosphere to limit the release of radioisotopes to the TSC that may leak from containment

5.1.1 Control Systems Evaluation and Instrument Setpoint Methodology

Reactor Protection System/Engineered Safety Features Actuation System Instrumentation Trip Setpoint and Allowable Values

The licensee in its submittal dated November 16, 2000, identified that instrument setpoints in the TS are established using the General Electric (GE) setpoint methodology. The staff has previously reviewed this instrument setpoint methodology and found it acceptable for establishing new setpoints in EPU applications. However, the staff was concerned about the use of a different setpoint methodology for BOP instruments. During a conference call on March 16, 2001, the staff requested the licensee to discuss the instrument setpoint methodology used for BOP instrumenation. The licensee in its response dated April 16, 2001, stated that they have not applied the GE setpoint methodology to non-safety-related instrumentation such as BOP instruments. The staff finds the licensee's response acceptable because this methodology is used for only non-safety related instrumentation and therefore does not require staff's review. During the conference call the staff also requested the licensee to add a reference of GE setpoint methodology in the bases section of the TS. The licensee in its response stated that the use of GE setpoint methodology is incorporated into the DAEC licensing basis via the UFSAR and the TS bases section provides a detailed discussion of the application of the methodology to explain the relationship between the analytical limits in the accident analysis and the development of the allowable values and trip setpoints. The staff finds the licensee's response acceptable.

The proposed setpoint changes resulting from the EPU are intended to maintain existing margins between operating conditions and the reactor trip setpoints and do not significantly increase the likelihood of a false trip or failure to trip upon demand. Therefore, the existing licensing basis is not affected by the setpoint changes to accommodate the EPU.

The following TS changes have been proposed by the licensee:

1. TS LCO 3.3.1.1: SR 3.3.1.1.2

   The licensee has reduced from 25 percent to 21.7 percent the percentage of reactor thermal-power (RTP) value, which is used to defer the surveillance requirement (SR) for 12 hours during plant startup. The RTP value being changed is contained in SR 3.3.1.1.2 and the associated note. The licensee's justification of this change is that the existing value is based on a point in the plant startup sequence where an accurate heat balance calculation can be performed by the plant computer which is tied to sufficient steam flow through turbine to synchronize the main generator to the grid. This steam flow, and in turn, reactor power level in Mwt are not being changed by the extended EPU and therefore, the percentage of RTP value is being revised to be consistent with the current absolute thermal power value. On this basis, the staff finds the proposed change to the TS acceptable.

2. TS LCO 3.3.1.1: Required Action E.1, SR 3.3.1.1.16, and Table 3.3.1.1-1 Functions 8 and 9

   The licensee has proposed to revise the percentage-of-RTP value corresponding to the power level where the reactor protection system (RPS) trip on turbine stop valve (TSV) or on turbine control valve (TCV) fast closure is automatically bypassed from 30 percent to 26 percent. The licensee's justification of this change is that these scram

signals are automatically bypassed at a low power level when the turbine bypass steam flow capacity is sufficient to mitigate a TSV or TCV closure transient. Because the turbine bypass capacity is not being changed by this EPU, the corresponding percentage of RTP is being revised to maintain the current thermal power value in Mwt corresponding to the existing bypass steam flow capacity. On this basis, the staff finds the licensee's justification for this TS change acceptable.

3. TS LCO 3.3.4.1: Applicability, Required Action C.2, and SR 3.3.4.1.4

   The licensee has proposed to reduce from 30 percent to 26 percent the percentage-of-RTP value corresponding to the power level where the end-of-cycle recirculation pump trip on TSV or TCV fast closure is automatically bypassed. The licensee's justification is that the revised value is consistent with the RPS trips discussed above, since this function is not required when the companion RPS functions are not required to be operable. On this basis, the staff finds the proposed change to the TS acceptable.

4. TS LCO 3.3.1.1: Table 3.3.1.1-1 Function 2b

   The licensee has proposed to revise the allowable value (AV) for the two-loop operation average power range monitor (APRM) flow-biased, high RPS trip with an equation for the AV. A footnote (c) is also being added to define the term "w" used in the AV equation. The licensee's EPU is based on the adoption of the maximum extended load line limit analysis (MELLLA). The licensee's safety analysis in the submittal is based on the MELLLA power/flow map and corresponding APRM AVs. The staff's review of the MELLLA is documented in other section of the safety evaluation prepared by the Reactor Systems Branch. Based on the acceptance of the MELLLA analysis, the staff finds the licensee proposed TS changes acceptable.

5. TS LCO 3.3.1.1: Table 3.3.1.1-1 Footnote b

   The licensee has proposed to replace the current AV for the single-loop operation APRM flow-biased high RPS trip with an equation for the AV. The new Footnote (c) identified above is used to define the term "w" used in the AV equation. The licensee has proposed this change to adjust the AV for the two-loop operation APRM flow-biased trip to account for the difference in recirculation-drive-flow to core-flow relationship in single loop operation. The higher core pressure drop associated with the EPU necessitates a different adjustment factor than the one currently used. Based on the acceptance of the MELLLA analysis, the staff finds the proposed change acceptable.

In addition to the above changes, the licensee will implement new setpoints for the instrumentation listed in the TS as percentage of flow or ~~pressure~~ power instrumentation without changing the percentage. The licensee has identified this instrumentation as follows:

(a) APRM scram fixed

(b) Main steam line high flow isolation

The anticipated transient without scram recirculation pump trip (ATWS-RPT) trips the pumps fc transients that result in an increase in reactor pressure and/or in low reactor water level. The ATWS analyses assume that the recirculation pumps will trip at the analytical limit of 1168.6 psig. The primary function of the ATWS-RPT is to reduce core flow, insert negative reactivity through the generation of voids, and reduce the core power during the initial part of the ATWS events. Therefore, the high-pressure ATWS-RPT setpoint is based on the value used in the ATWS analysis. The licensee stated that the ATWS-RPT low reactor level setpoint is not a significant factor for the limiting analyzed ATWS events.

The licensee used the current high-pressure ATWS-RPT setpoint in the EPU ATWS evaluation in Section 9.3 and stated that the calculated peak vessel pressure remains below the ASME upset limit of 1500 psig. The licensee concluded that the current EPU high-pressure ATWS-RPT setpoint is acceptable for the EPU and remains unchanged as indicated in Table 5-1. Since the EPU ATWS analyses demonstrated that the peak vessel pressure will remain below the ASME upset limit, assuming the current setpoint of 1168.6 psig, as stated by the licensee, the staff accepts that the limit need not change.

5.1.3.3 Safety/Relief Valve

The licensee will not change the SRV and SV analytical limits, since the reactor operating pressure will not change, and the ASME overpressure protection and pressurization transients are based on the current setpoints. The staff accepts this.

5.1.3.4 Neutron Monitoring System

The licensee will not change the analytical limit (AL) (as percentage of RTP) for the fixed APRM high power scram. Therefore, the licensee will maintain the values in the TS for the allowable value and the nominal trip setpoints. This is consistent with Section F.4.2.2 of ELTR1. The licensee also evaluated all of the limiting transients that rely on the fixed APRM trip at the EPU conditions.

Since the EPU operation will include implementing the MELLLA region, the licensee developed new equations for the flow-biased APRM scrams, both for single and two recirculation loop operation. The licensee stated that that design bases for the MELLLA operating regime uses a linear relationship for all analytical limits versus the recirculation drive flow, which is consistent with the APRM hardware design and licensing analyses. According to the licensee, the ALs for the flow-biased APRM scrams are straight line equations, in which the slope was changed consistent with other BWR MELLLA applications. The licensee also maintained equivalent margins between the rod blocks and scram trip setpoints to avoid spurious protective actions. The flow-biased APRM scram analytical limits are also specified in Table 5-1 of the submittal.

The flow-biased APRM scram setpoint also depends on the requirement of the Option I-D long-term stability solution to provide adequate MCPR safety limit protection from core wide T/H oscillation. The staff agrees with the licensee's assessment.

The RBM limits erroneous rod withdrawals by supplying a trip signal to the Reactor Manual Control System to block further withdrawal. The trip is initiated when the RBM output exceeds the rod block setpoint. The licensee stated that the setpoint has not changed for the three upscale rod block trip levels. Because the limiting transients which rely on these setpoints have

(in terms of percent core thermal power)

these

been reanalyzed to ensure that the existing setpoint provides adequate protection, the staff finds that the use of the existing setpoints is acceptable.

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

6.1 AC Power

6.1.1 Offsite Power System

The staff has reviewed information provided by the licensee to determine the impact of the EPU on offsite power. The areas reviewed were the grid stability analysis and the analysis of related electrical systems.

6.1.1.1 Grid Stability and Reliability Analysis

A grid stability analysis was performed for 1790 MWt (641 MWe - gross) to demonstrate conformance to GDC 17 of 10 CFR Part 50, Appendix A. GDC 17 addresses onsite and offsite electrical supply and distribution systems for safety-related components. Operation at 1790 MWt (EPU Phase I) has no significant effect on grid stability or reliability. This grid stability analysis was discussed in the DAEC EPU safety evaluation report. The current output capability is based on the currently installed main transformer (with upgraded cooling units) rating of 660 MVA and the isolated phase bus rating of 18000 amperes.

(Reference 19)

In response to a staff's request for additional information on the grid stability analysis for the EPU of 1912 MWt, the licensee stated that this analysis will be reperformed when the main transformers are replaced. Many factors directly affect the grid stability analysis. Proposed additions that would affect the current grid stability analysis include a 345 kV transmission line from the Arrowhead substation in Minnesota to the Weston substation in Wisconsin, the potential installation of gas turbine generation capacity within the Alliant Energy territory, and an additional 2000 MWe of new generation in the Alliant territories in Iowa and Wisconsin in the near future. The licensee plans to update the existing grid stability study as additional changes occur, as required by its licensing basis in UFSAR Section 8.2.2.1. The licensee believes that an analysis performed today, projecting the conditions at the time DAEC achieves the EPU of 1912 MWt, would not demonstrate that grid stability will remain acceptable. The licensee committed to perform a new stability analysis for the main transformer replacement prior to increasing power above the 1790 MWt level and for any significant increase in actual plant electrical output from operation at the EPU level of 1912 MWt (EPU phase II).

This is a misquote from Ref. 19 what we said was: "We believe that an analysis performed today, projecting the conditions at the time DAEC achieves the power uprate of 1912 MWt, would not be meaningful in demonstrating that grid stability will remain acceptable." (emphasis added).

In response to the staff's request for additional information, the licensee described the assumptions, methods, inputs, results, findings, and conclusions of the current grid stability analysis for Phase I.

The licensee determined that no modifications were required for the generator breakers. Additionally, a 40 MVAR capacitor bank will be installed in the switchyard in the fall of 2001 to partially compensate for the loss of reactive power capability of the main generator as a result of the EPU.

Note: We did not make this specific commitment. Per Reference 19, we stated: "The UFSAR commits us to performing a new stability study for the Main Transformer replacement, which is required to increase power above the 1790 MWt (641 MWe) level and for any significant increase..."

On the basis of this information, the staff concludes that the proposed EPU to 1790 MWt at DAEC will not adversely affect the grid stability and reliability. The licensee has committed to perform a new stability analysis as required in the DAEC UFSAR before increasing the power above 1790 MWt.

6.1.1.2 Related Electrical Systems

The licensee performed a EPU review to determine the adequacy of electrical systems associated with the main turbine-generator auxiliary systems.

6.1.1.2.1 Main Generator

The existing main generator is rated at 715.225 MVA, 0.95 power factor, and 22 kV. The expected generator output is 679.46 MW at 0.95 power factor. The net plant EPU-related output will be 677 MWe and is within the capability of the generator. The licensee plans to replace the hydrogen cooling units to improve heat removal capability to accommodate full uprated power production during Phase I. The review determined that the electrical system's configuration and operating voltage ranges would be unchanged and would remain adequate for operation at the higher output.

6.1.1.2.2 Isolated Phase Bus Duct

The existing isolated phase bus duct rating is 18000 amps and 22 kV for the main section. The maximum current output is 18232 amperes [660000/(1.7321x22x0.95)] with a main transformer output of 660 MVA and 95 percent of 22 kV. The review determined that the isolated phase bus duct would be adequate for both rated voltage and low-voltage current output for EPU Phase I. The maximum current output is 19757 amps [715.225MVA/(1.7321x22x0.95)] with a generator output of 715.225 MVA and 95 percent of 22 kV. The licensee proposes to modify the isolated phase bus to increase its rating from 18000 to 20000 amperes and improve the bus cooling unit's heat removal capacity during Phase II.

6.1.1.2.3 Main Transformer

The existing main transformer rating is 600 MVA and 22/161 kV. The licensee decided to add new oil-cooling units to increase the transformer's rating from 600 MVA to 660 MVA. This will be done during EPU Phase I. The existing main transformer must be replaced to accommodate the full EPU power output from the main generator. This will be done during EPU Phase II.

Thus the turbine-generator and major electrical components from the isolated phase bus to the switchyard will remain adequate for operation at the higher output after the proposed modifications.

6.1.2 Onsite Power Distribution System

The onsite power distribution system consists of transformers, buses, switchgear, and distribution panels. The alternate current (ac) power to the distribution system is provided from

On the basis of this information, the staff concludes that the proposed EPU at Duane Arnold has no impact on the dc power system.

6.3 Fuel Pool Cooling

The licensee plans to pursue advanced core designs including the use of GE-14 fuels for the DAEC. Accordingly, in a separate licensing amendment request (dated November 17, 2000), the licensee presented an SFP thermal-hydraulic analysis to reflect the effects of GE-14 fuels used in the advanced core designs that increase fuel burnup, cycle length, and reload batch size on SFP cooling. Also, the SFP thermal-hydraulic analysis was performed for a power level of 1950 MWt (102 percent of 1912 MWt) with the anticipation of the DAEC operations at the proposed EPU level.

In a SE dated August, 2001, the staff found that the SFP thermal-hydraulic analysis complies with the intent of the guidance described in SRP Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," for SFP, and is acceptable. Therefore, the staff concludes that the design and operation of the SFP cooling systems (including SFP cooling system and the RHR system in the SFP cooling assist mode) at the DAEC are acceptable for plant operations at the proposed EPU level.

6.4 Water Systems

6.4.1 Service Water Systems

The service water systems are designed to provide cooling water to various systems (both safety-related and non-safety-related systems).

6.4.1.1 Safety-Related Service Water Systems (Safety-Related Loads)

These systems include the emergency service water (ESW) system and the residual heat removal service water (RHRSW) system. All heat removed by these systems is rejected to the ultimate heat sink.

6.4.1.1.1 Emergency Service Water System

The ESW system provides cooling water to the following essential components/systems following a LOCA: emergency diesel generator coolers, RHR pump seal coolers, RHR and core spray pump room cooling units, HPCI and RCIC room cooling units, control building chillers, core spray and RHRSW pump ~~bearing~~ motor coolers, heating ventilating and air conditioning (HVAC), instrument air compressors, and SFP emergency makeup (if needed).

The licensee performed evaluations and stated that the performance of the ESW system during and following a LOCA is not significantly dependent on the RTP. The heat loads from diesel generators remain unchanged for LOCA conditions following uprated operations. The building cooling loads also remain the same because equipment performance in these areas remains essentially unchanged for post-LOCA conditions. In addition, the ability to supply emergency

makeup[2] to the SFP is not changed following the EPU condition. Therefore, plant operations at the proposed EPU do not require the modification of the ESW system for the safety-related loads.

Based on our review of the licensee's evaluation and rationale, and the experience gained from our review of EPU applications for other BWR plants, we find that DAEC operations at the proposed EPU do not change the design aspects and operations of the ESW system, and have an insignificant or no impact on the ESW system. Therefore, we conclude that the ESW system at DAEC remains adequate for plant operations at the proposed EPU to perform its safety function during and following a LOCA.

6.4.1.1.2 Residual Heat Removal Service Water System

The RHRSW system provides cooling water to the RHR heat exchangers[3] under normal or post-accident conditions, and is capable of supplying water to flood containment for post-accident recovery. The licensee performed containment pressure and temperature response analyses which demonstrate that the capability of the containment system is adequate to operate at the proposed EPU. In the containment pressure and temperature response analyses, the post-LOCA RHRSW flow rate and temperature were assumed to be unchanged for EPU conditions. Therefore, the RHRSW system remains adequate for plant operations at the proposed EPU to perform its safety function during and following a LOCA. The staff's evaluation of the containment system performance for plant operations at the proposed EPU is addressed in Section 4.1.

During shutdown cooling with the RHR system, heat loads on the RHR heat exchangers will increase proportionally to the increase in reactor operating power level, thus increasing the time required to reach the shutdown temperature. The licensee stated that this has no effect on plant safety. The staff's evaluation of the effect of plant operations at the proposed EPU on shutdown cooling with the RHR system is addressed in Section 3.9.1.

Section 3.9.1 does not discuss shutdown cooling (or LPCI).

In addition, since DAEC operations at the proposed EPU do not change the design aspects and operations of the RHRSW system, the capability of the RHRSW system to flood the containment following a LOCA is not impacted by plant operations at the proposed EPU.

Based on our review of the licensee's rationale, the evaluation described above, and the experience gained from our review of EPU applications for other BWR plants, we find that the RHRSW system is acceptable for DAEC operations at the proposed EPU.

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[2] As reported in the SE cited in Section 6.3 above regarding reracked DAEC spent fuel pool, the calculated maximum boiloff rate is 53.05 gpm, which is less than the design makeup capacity of 59.5 gpm available from the ESW system.

[3] The long-term containment pressure and temperature responses following a LOCA are governed by the ability of the RHR system to remove the decay heat from the suppression pool.

6.4.1.2 General Service Water System (Non-Safety-Related Loads)

The general service water (GSW) heat loads will increase approximately proportional to the increase in the reactor operating power level. The licensee stated that the major GSW heat load increases from the EPU reflect an increase in main generator heat losses rejected to the stator water coolers, hydrogen coolers and exciter coolers in addition to increases in bus cooler heat loads. The licensee performed evaluations which demonstrate that the GSW system is adequate for plant operations at the proposed EPU.

Since the GSW system does not perform any safety-related function, the impact of the proposed EPU on the designs and performances of this system was not reviewed.

6.4.2 Main Condenser, Circulating Water, and Normal Heat Sink System Performance

The main condenser, circulating water, and normal heat sink systems are designed to provide the main condenser with a continuous supply of cooling water for removing heat rejected to the condenser, thereby maintaining condenser pressure as recommended by the turbine vendor. The licensee stated that the performance of the main condenser, circulating water, and normal heat sink systems was evaluated and found adequate for plant operations at the proposed EPU.

Since the main condenser, circulating water, and normal heat sink systems do not perform any safety-related function, the impact of the proposed EPU on the designs and performances of these systems was not reviewed.

6.4.3 Reactor Building Closed Cooling Water System

The reactor building closed cooling water (RBCCW) system is designed to remove heat from various auxiliary plant equipment housed in the reactor building during normal plant operations. The licensee performed evaluations and stated that the increase in heat loads on this system due to plant operations at the proposed EPU is insignificant.

Since the RBCCW system does not perform any safety-related function, the impact of the proposed EPU on the designs and performances of this system was not reviewed.

6.4.4 Well Water System

The well water (WW) system supplies cooling water to many of the non-safety HVAC units. The licensee performed evaluations and stated that the WW system heat loads do not increase significantly due to the EPU. The WW system has adequate heat removal capability for plant operations at the proposed EPU.

Since the WW system does not perform any safety related function, the impact of the proposed EPU on the designs and performances of this system was not reviewed.

6.4.5 Ultimate Heat Sink (UHS)

By letter dated June 21, 2001, in response to the staff's RAI, the licensee explained that for the evaluation of the turbine building HVAC performance, assumptions were made to maximize the duty on the cooling units. For example, the largest increase in the "sub-area" heat load (i.e., an increase of 21 percent near the main condenser and feedwater heater #1) was conservatively assumed to apply to the entire condenser/heater bay area. Most "sub-areas" were predicted to experience an increase in heat load of only 4 percent to 10 percent.

The licensee further explained that the turbine building HVAC system is sized with enough margin to accommodate the expected increase in actual heat load for the feedwater heater bay area. This can be seen by the predicted small increase in outlet cooling water temperature of the coolers for the above increase (21 percent) in heat load. Because of the excess in cooling capacity and the use of very conservative inputs to the analysis, the licensee does not expect to see any noticeable increase in actual operating temperatures in this area.

Based on our review of the licensee's rationale, and the experience gained from our review of EPU applications for other BWR plants, we conclude that the EPU does not adversely affect the operation of HVAC.

6.7 Fire Protection Program

We find that the operation of the plant at the EPU RTP will have no impact on the existing fire detection or suppression systems, the existing fire barriers provided to protect safe-shutdown capability, or the administrative controls that are specified in the plant's fire protection plan required by 10 CFR 50.48(a). The NRC requirements for achieving and maintaining safe shutdown following a fire require (1) that one train of systems necessary to achieve and maintain hot shutdown be maintained free of fire damage, and (2) that the systems necessary to achieve and maintain cold shutdown can either be repaired within 72 hours if using redundant systems, or the systems can be repaired and cold shutdown can be achieved within 72 hours if using alternative or dedicated shutdown capability. While Section 6.7 "Fire Protection," of the submittal report only addresses cold shutdown capability and is silent concerning hot shutdown capability, Table 6-3 of the submittal indicates that the limits for important reactor process variables (peak cladding temperature, primary systems pressure, primary containment pressure, and suppression pool bulk temperature) are not exceeded following a fire event.

Actually section 6.7.1 states that one train of systems is available to achieve and maintain "safe shutdown conditions." It goes on to state "The safe shutdown system can achieve cold shutdown conditions within 72 hours." Our use of the term "safe shutdown" is meant to imply "hot shutdown."

Section 3.8 "Reactor Core Isolation Cooling," states that for certain beyond-design-basis events (Appendix R or ATWS), operation of the RCIC system at suppression pool temperatures greater than the operation limit may be accomplished by using the dedicated condensate storage tank (CST) volume source of water. Section 4.2.1 "High Pressure Coolant Injection," states that for certain beyond-design-basis events (Appendix R or ATWS), operation of the HPCI system at suppression pool temperatures greater than the operation limit may be accomplished by using the dedicated CST volume source of water. While the staff does not consider an Appendix R fire a beyond-design-basis event, as stated in the submittal report, the staff has accepted the use of either HPCI or RCIC with suction from either the suppression pool or the CST for providing reactor coolant makeup to achieve hot shutdown when those systems are protected in accordance with the requirements specified in Section III.G of Appendix R to 10 CFR Part 50.

This is the DAEC's licensing basis. We were constructed prior to the promulgation of App. R., thus it is not a "design basis event" for the DAEC. It was a retrofit.

10 CFR Part 50, Appendix I, include the offgas system and SGTS, as well as other building ventilation systems. Various devices and processes, such as radiation monitors, filters, isolation dampers, and fans, are used to control airborne radioactive gases. The results of the licensee's analyses demonstrate that airborne effluent activity released through building vents is not expected to increase significantly due to plant operations at the proposed uprate power level. The release limit is an administratively controlled variable, and is not a function of core power.

Based on our review of the licensee's rationale and evaluation, and the experience gained from our review of EPU applications for BWR plants, the staff concludes that plant operations at the proposed uprate power level will have an insignificant impact on the above systems.

8.2.1 Offgas System

Core radiolysis (i.e., formation of H_2 and O_2) increases linearly with core power, thus increasing the heat load on the offgas recombiner and related components. The licensee evaluated the impact of the increases of these offgases resulting from plant operation at the proposed EPU on the offgas system. The licensee stated these operational increase in offgas due to EPU remains well within the design capacity of the system. The system radiological release rate is administratively controlled, and does not change with operating power. Therefore, EPU does not affect the offgas system design or operation.

Based on our review of the licensee's rationale and evaluation, and the experience gained from our review of EPU applications for other BWR plants, the staff concludes that plant operations at the proposed EPU will have an insignificant impact on the offgas systems.

Note: No discussion of PUSAR Sections 8.3-8.5 on Radiation Sources?

9.0 REACTOR SAFETY PERFORMANCE FEATURES

9.1 Reactor Transients

AOOs are abnormal transients which are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, GDC 10, 15, and 20. GDC 10 requires that the reactor core and associated control and instrumentation systems be designed with sufficient margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation and during AOOs. GDC 15 stipulates that sufficient margin be included to ensure that the design conditions of the RCPB are not exceeded during normal operating conditions and AOOs. GDC 20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure that the specified fuel design limits are not exceeded during any normal operating condition and AOOs.

The SRP further states (1) pressure in the reactor coolant and main steam system should be maintained below 110 percent of the design values according to the ASME Code, Section III, Article NB-7000, "Overpressure Protection"; (2) fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during normal operating conditions and AOOs; (3) an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and (4) an

incident of moderate frequency, in combination with any single-active component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel cladding perforations are acceptable.

The DAEC UFSAR evaluates a wide range of potential transients. Chapter 15 of the UFSAR contains the design basis analyses that evaluate the effects of an AOO resulting from changes in the system parameters such as (1) a decrease in core coolant temperature, (2) an increase in reactor pressure, (3) a decrease in reactor core coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, (6) a decrease in reactor coolant inventory. The plant's responses to the most limiting transients are analyzed each reload cycle and are used to establish the thermal limits. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

The generic guidelines for EPU evaluation (Appendix E of ELTR1) identified (a) the limiting transient to be considered in each event category, (b) the analytical methods to be used, (c) the operating conditions assumed in the generic evaluation presented in the report, and (d) the criteria that was applied. The licensee stated that in support of the EPU, each limiting transient analysis for each category of the transients listed in Table E-1 of ELTR1 was analyzed. Table 9-1 of the submittal describes the reactor operating conditions used in analyzing the limiting transients for the recent pre-EPU fuel cycle (Reload 16) and for the EPU representative core. The table also lists the nominal dome operating pressure and the SLMCPR used in the transient analyses and in calculating the MCPR operating limits. The EPU transients analyses were based on a representative GE-14 core and the licensee calculated an SLMCPR value of 1.08 for the core.

The licensee stated that input parameters related to the performance improvement features or equipment out of service (OOS) have been included in the safety analyses for the EPU. DAEC is licensed or seeks to implement MELLLA, SLO, one ADS OOS, ARTS, turbine bypass valve (TBV) OOS, recirculation pump trip (RPT) OOS, and MSIV OOS for EPU operation. Therefore, the EPU transient analyses that were performed considered these operating features. According to the licensee, most of the transient events are analyzed at full power and at the maximum allowed core flow operating point on the power/flow map (Figure 2-1). The licensee also included the 2 percent power uncertainty in the analyses. The licensee analyzed the following limiting transients and Table 9-2 of the submittal provides the results.

More accurately add: "..., except where it is already accounted for in the analysis methodology or not required." (per PUSAR 1.2.1)

- load rejection with bypass failure (LRWOB)
- turbine trip with bypass failure (TTWOB)
- feedwater controller failure (FWCF) maximum demand
- loss of feedwater heating (LFWH)
- inadvertent HPCI actuation
- rod withdrawal error (RWE)
- fast recirculation increase

- slow recirculation increase
- load reject with bypass
- MSIVC all valves
- MSIVC one valve

The licensee determined that, as shown in Table 9-2 and Figures 9-1 to 9-4 of Reference 1, there are no changes to the basic characteristics of any of the limiting events due to EPU operating conditions.

Table 9-2 does not include the pressure regulator downscale failure transient, which is included in Table E-1 of ELTR1. However, the plant is equipped with a backup regulator and this event is considered to be a mild transient. Also, other transients in the category of decrease in heat removal by the reactor coolant system bound this event. Also Table 9-2 does not include LOFW, which forms the basis for evaluating the RCIC capability to perform its design basis function. As discussed in Section 3.8 of this document, the licensee analyzed this transient. Appendix E of ELTR1 does include a loss of single feedwater pump transient, but the licensee considers this event to be nonlimiting with respect to MCPR and did not analyze it. The staff agrees with the licensee's assessment. Incorrect. See PUSAR 7.4.2.

In its February 8, 1996, response to ELTR1, the staff agreed that the minimum set of limiting transients described in Appendix E of the topical needed to be included in the uprate amendment request. The staff also stated that a list of all of the transients analyzed in support of the EPU should be included, with an explanation of how the limiting transients were selected. The submittal did not provide the bases for selecting the EPU limiting transients. GE selects the limiting EPU transients by evaluating the seven categories of transient events based on the EPU parameters to ensure that (a) the UFSAR events remain bounded by the reload transient events, (b) no nonlimiting events become limiting in terms of thermal limits due to the EPU, and (c) no additional limiting event in terms of thermal limits are caused by the EPU operating conditions. Appendix E.2.2 of ELTR1 also discusses the bases for selecting the limiting transients to analyze in support of the EPU. The stated justifications are applicable to DAEC.

In support of operation at the higher MELLLA rod line and the EPU power level, the licensee analyzed the limiting transients using representative GE-14 core. The current EPU analyses are based on NRC-approved analytical methods and codes. The transient evaluations also take into account the impact of the performance improvement programs or special features in establishing the thermal limits for the EPU operation. Therefore, staff agrees with the licensee's conclusions that the EPU transient analyses do not identify any major changes to the basic characteristics of any of the limiting events due to the EPU operating conditions. The licensee will be uprating in phases, and will analyze or confirm the limiting transients based on the actual core design as described in Section 1.3 and the uprated power level at each phase of the uprate process. The staff finds this approach acceptable.

In the current TS, some limiting conditions for operation (LCOs) and surveillance requirements (SR) use 25 percent of the RTP to determine when to apply the corresponding

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requirement. The value of 25 percent of RTP is based on the steam flow capability of the bypass valves. The limiting fast transients assume that several scram signals are bypassed below 25 percent of rated thermal power. The licensee intends to reduce the 25 percent RTP limit to 21.7 percent of the EPU power level. Therefore, the actual thermal power thresholds used in TS LCO and SR requirements will remain at the same value of 414.5 MWt, which represents 25 percent of the pre-EPU thermal power and 21.7 percent of the EPU power level, and represents the steam flow capacity of the bypass valves. The staff finds this acceptable.

9.2 Design Basis Accidents

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, "Accident Source Term," which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their design-basis accident (DBA) analyses with alternative source term (ASTs). Regulatory guidance for the implementation of these ASTs is provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Powe Reactors." Section 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs. NMC's application of October 19, 2000, addresses these requirements in proposing to use the AST described in RG 1.183 as the DAEC DBA source term used to evaluate the radiological consequences of DBAs. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC)-19, for the loss-of-coolant accident (LOCA), the main steamline break (MSLB) accident, and the control rod drop accident (CRDA).

The accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large LOCA. As a result of significant core damage, fission products are available for release into the containment environment. An AST is an accident source term that is different from the acciden source term used in the original design and licensing of the facility and has been approved for use under 10 CFR 50.67. Although an acceptable AST is not set forth in the regulations, RG 1.183 identifies an AST that is acceptable to the staff for use at operating reactors.

By application dated October 19, 2000, as supplemented March 23, April 9, and June 27, 2001, the licensee requested a license amendment for the Duane Arnold Energy Center (DAEC). Amendment No. 240 was granted on July 31, 2001. The amendment allows the replacement the accident source term used in design basis radiological analyses with an alternative source term (AST) pursuant to 10 CFR 50.67. The NRC had previously approved the TS changes associated with secondary containment operability during refueling operations and the selective implementation of the AST to the FHA in Amendment No. 237, dated April 16, 2001. The SE dated July 31, 2001, addresses the staff review of the DBA analyses. The staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. In doing this review, the staff relied upon information placed on the docket by licensee, staff experience in doing similar reviews and, where deemed necessary, on staff confirmatory calculations. The staff concluded that the licensee used analysis methods and assumptions consistent with the conservative guidance of RG 1.183, the proposed TS changes, and the future EPU.

limiting ATWS events in terms of overpressure and suppression pool cooling: (a) MSIVC, (b) Pressure Regulator Failure-Open (PRFO), (c) LOOP, and (4) inadvertent opening of a relief valve (IORV). The licensee performed the ATWS analyses, as discussed in ELTR1, at the MELLLA/EPU operating condition to demonstrate that DAEC meets the ATWS acceptance criteria. To provide a benchmark for the plant's response to limiting ATWS events, the licensee also performed the ATWS analyses based on the ORTP.

Tables 9-7 and 9-8 of the submittal provide the key input parameters used in the ATWS analyses and the corresponding results (peak vessel bottom pressure, peak cladding temperature, peak suppression pool temperature, and peak containment pressure). The licensee stated that the results of the ATWS analyses meet the ATWS acceptance criteria. Therefore, the plant's response to an ATWS event for EPU operation is acceptable.

Table 9-8 shows that the ATWS PCT for the ORTP is 1418 °F and the EPU PCT is 1380 °F. The staff asked the licensee to verify the PCT values. The licensee confirmed that the PCT values are correct and explained the bases for these values, which seem to be counterintuitive. The staff also found similar trends for other calculations (pre-EPU PCTs are higher than the EPU PCTs). Since the ATWS analyses are based on NRC-approved methods and the licensee performed the ATWS analyses at the MELLLA/EPU conditions, the staff accepts the licensee's evaluation.

The staff agrees with the licensee that DAEC meets ATWS mitigating features stipulated in 10 CR 50.62 and the results of the ATWS analyses at the EPU/MELLLA operation meet the ATWS acceptance criteria. Future reload analyses would confirm that the plant's response to an ATWS event based on the cycle-specific condition will continue to meet the ATWS acceptance criteria.

ATWS-RPT Instability

during an ATWS event.

The EPU submittal did not address whether operation at the higher MELLLA/EPU condition might affect the potential for T/H instability. The potential for instability following a transient ATWS-RPT event has been the subject of extensive research and discussions between the NRC and the BWROG. In NEDO-32047-A, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," the BWROG and GE generically addressed the potential for T/H instability during an ATWS event and its impact on high-density BWR/5 and BWR/6 reactors operating at reactor power levels of 3323 MWt. The staff reviewed the topical report and issued an SE on February 5, 1994. In the SE, the staff stated :

(a) Although large power oscillations may worsen the overheating and severity of fuel damage resulting from ATWS event, the analyses indicate that core coolability and containment integrity can be maintained. The staff concluded that the prescriptive requirements of the ATWS rule remained appropriate.

(b) The BWROG had proposed to revise the EPGs and direct immediate action to be taken by the operators to reduce core inlet subcooling after confirmation of an ATWS event and direct earlier injection of boron in the presence of power oscillations. The staff noted that the risk prospective

reviews of the proposed actions were continuing, but these actions are sufficient for mitigating the consequences of a bounding ATWS event with oscillation.

(c) By reducing the core inlet subcooling, the EPGs will instruct the operators to continue decreasing the water level and maintain it at a level of about 1 meter or more below the feedwater spargers to ensure effective termination of power oscillation. Reducing the water level below the feedwater spargers would ensure that when the feedwater flow is resumed, the incoming cold water is heated by mixing with the steam. The staff stated that injection of additional water would not provide significant benefit to the response to instability and other criteria must be used to determine whether water level reduction below the TAF is warranted.

Instability in ATWS

The staff accepted NEDO-32047 and NEDO32164, "Mitigation of BWR Core Thermal-Hydraulic ~~Stability~~," in a letter dated February 5, 1994. The staff stated that the recommended operator actions to lower water level to below the feedwater nozzles and earlier SLC system activation were appropriate for mitigating the ATWS event with oscillations.

Section L.3.1, "Power Conditions for ATWS Evaluation," and Section L3.2, "Operator Action," o ELTR1 discuss some aspects of ATWS instability. The EPU report stated that the effect of operation along the maximum extended operating domain (MEOD) will be considered, explaining that MEOD operation will maximize the natural circulation power level after an ATWS-RPT. In addition, ELTR1 stated that operator actions would be assumed to be consistent with the BWR EPGs, with typical operator actions for ATWS including:

(1) tripping the feedwater pumps on high suppression pool temperature or other confirmed ATWS symptoms,
(2) starting the SLC system on confirmed ATWS symptoms,
(3) maintaining the RPV water level near the TAF during an ATWS event, and
(4) starting the RHR in the pool cooling mode on high suppression pool temperature.

The EPU topical further stated that in "some areas, manual actions are involved in ATWS evaluation. It is most consistent for the plants to assume that these actions are performed in response to symptoms as they may occur during the postulated event." Therefore, even though ELTR1 did not directly discuss ATWS instability, it indirectly pointed out that, for plants operating along the high MEOD operating domain, an ATWS-RPT event would place the reactor in an instability zone with high power at the natural circulation conditions. It also recommended tripping the feedwater pumps to reduce subcooling, which is an important destabilizing factor. Maintaining the reactor water level at the TAF (below the feedwater spargers) would also reduce subcooling and excess reactivity.

In the absence of a plant-specific evaluation of the impact of the higher MELLLA rod line and power density on DAEC's ATWS instability response, the staff asked the licensee to confirm that DAEC's EPGs are consistent with ELTR1 and the recommendations in NEDO-32047-A. In its August 16, 2001, submittal, the licensee confirmed that the basic strategies and operator responses in the Emergency Operating Procedures (EOPs) have not been changed as a result

In SAIC-91/6670, "Technical Evaluation Report, Duane Arnold Energy Center, Station Blackout Evaluation," Science Application International Corporation (SAIC - an NRC contractor) reviewed DAEC's SBO response and coping capability at the pre-EPU power level. The technical evaluation report (TER) stated that the licensee maintains 75,000 gallons of condensate inventory and that 62,800 gallons of condensate would be needed for decay heat removal and cooldown during a 4-hour event. According to the pre-EPU TER, the assumptions and input parameters are consistent with the expected conditions and therefore, the site has sufficient condensate to cope with an SBO event. The pre-EPU TER also reviewed DAEC's ability to maintain adequate reactor core coolant and inventory during an SBO event. The pre-EPU SBO analysis assumes that the SRVs lift early in the event, and HPCI and RCIC, which rely on steam driven turbines, initiate and inject into the reactor. HPCI and RCIC stop on high reactor vessel water level and the operators subsequently prevent injection of the ~~high coolant~~ HPCI systems, operating the HPCI in the recirculation mode to and from the CST. Water level is maintained using the RCIC and 30 minutes into the event, the operators start controlled reactor vessel depressurization at less than the 100 °F/hr limit. At 100 minutes, the RPV is maintained at about 200 psig, using HPCI and RCIC as necessary.

The higher decay heat for the EPU operation would increase the boil-off rate, therefore, the ability of the plant to maintain core coverage, using the available inventory in the CST would be affected. In its August 16, 2001, response to a staff RAI, the licensee stated that its new 4-hour coping analysis determined that approximately 66,750 gallons of CST inventory would be used, and this continues to be less than the 75,000 gallons available in the CST.

The staff has reviewed DAEC's ability to cope in a 4-hour duration SBO and ensure core cooling and coverage during the event. The staff accepts the licensee's conclusion that the plant's SBO coping capabilities will not be adversely affected by EPU operation.

Based on our review and the experience gained from our review of EPU applications for other BWR plants, we agree with the licensee's conclusion that the impact of plant operations at the proposed EPU on the systems and equipment used to cope with an SBO event is insignificant. Further, the NRC staff has reviewed the information provided by the licensee to determine the impact of the EPU on the existing analysis for SBO at DAEC and agree with the licensee's rationale and evaluation that the plant continues to meet 10 CFR 50.63 requirements.

10. ADDITIONAL ASPECTS OF EXTENDED EPU

10.1 High-Energy Line Breaks

To achieve the proposed higher power at the DAEC, the licensee plans to expand the operating envelope on the power/flow map through implementation of maximum load line limit analysis (MELLLA). Operation at the EPU level does not require an increase in the reactor vessel dome pressure over the pre-EPU value to supply more steam to the turbine. Therefore, plant operations at the EPU level will have an insignificant impact (due to changes in the fluid conditions, i.e., pressure or enthalpy, within the system piping) on the mass and energy release rates following a high-energy line break (HELB) outside the primary containment.

10.1.1 Temperature, Pressure and Humidity Profiles Resulting From HELB

The licensee performed an HELB analysis for all systems (e.g. main steam system, feedwater system, reactor core isolation cooling system, etc.) evaluated in the USAR. The licensee statec that the resulting environmental conditions (i.e., pressure, temperature, and humidity profiles) due to plant operations at the proposed EPU level are bounded by the existing profiles used to qualify equipment and systems that support a safety-related function.

Based on our review of the licensee's analysis, the staff agrees with the licensee that the existing environmental conditions used to qualify equipment and systems that support a safety-related function remain bounding for the pressure, temperature, and humidity profiles resulting from an HELB outside the containment and are acceptable for plant operations at the proposed EPU level.

10.1.1.1 Main SteamLine Break

The licensee stated that the critical parameter normally affecting the MSLB analysis relative to the EPU would be an increase in reactor vessel dome pressure. Since there is no increase in the reactor vessel dome pressure, there is no increase in the blowdown rate following an MSLB in the steam tunnel. Therefore, the pressure and temperature profiles following an MSLB in the steam tunnel are not affected for plant operations at the proposed EPU level.

Based on our review of the licensee's evaluation and rationale, the NRC staff agrees with the licensee's conclusion that the existing pressure and temperature profiles following an MSLB in the steam tunnel are not affected and are acceptable for plant operations at the proposed EPU level.

10.1.1.2 Feedwater Line Break

At the EPU level, the feedwater temperature, pressure, and flow rate increase slightly. The licensee performed an analysis for a feedwater line break in the steam tunnel. The licensee stated that design margins within the HELB analysis for a feedwater line break in the steam tunnel are conservative and are bounded by the MSLB.

Based on our review of the licensee's rationale and evaluation, the NRC staff agrees with the licensee that the pressure and temperature profiles following a feedwater line break in the steam tunnel are bounded by the MSLB.

10.1.1.3 High-Pressure Coolant Injection Line

Because there is no increase in the reactor dome pressure relative to the current analyses, the mass release rate following a HPCI or RCIC line break does not increase. The licensee stated that the previous analyses for these line breaks are bounding for the EPU conditions.

Based on our review of the licensee's rationale, the NRC staff agrees with the licensee that the previous analyses for these line breaks remain bounding for the EPU conditions.

remains qualified because its as-tested qualification for total integrated dose (TID) exceeds the EPU values. For example, for the RCWU heat exchanger room, which has a 40-percent increase in normal dose, the calculated TID is 8.34E+06 rads. The minimum as-tested TID for equipment in that room is 2.0E+07 rads. Thus, the EPU total dose (i.e., normal and accident) in these rooms is bounded by the current dose level used to qualify all components potentially affected by this increase. Therefore, these components continue to be qualified for the EPU.

In summary, the EPU has a negligible effect on the environmental conditions currently used by the EQ program for safety-related electrical equipment outside the primary containment.

10.2.2. EQ of Mechanical Equipment with Nonmetallic Components

In response to a staff request for additional information, the licensee stated that the DAEC plant design control program ensures that nonmetallic components (i.e., seals, gaskets, lubricants, and diaphragms) are properly specified and procured for the environment in which they are intended to function.

Based on our review of the licensee's rationale, and since the changes for the normal and accident environmental conditions inside and outside the containment and for the system process temperatures are negligible,the staff conclude that the EQ of the nonmetallic components exposed to the EPU conditions is not adversely impacted.

10.2.3 Mechanical Components Design Qualification

10.2.3.1 Equipment Seismic and Dynamic Qualification

The licensee evaluated equipment qualification for the EPU condition. The dynamic loads such as SRV discharge and LOCA loads (including pool swell, condensation oscillation, and chugging loads) that were used in the equipment design will remain unchanged as discussed in Section 4.1.2 of Reference 2. This is because the plant-specific hydrodynamic loads which are based on the range of test conditions for the design-basis analysis at DAEC, are bounding for the EPU condition. 1

Based on its review of the proposed EPU amendment, the staff finds that the original seismic and dynamic qualification of safety-related mechanical and electrical equipment is not affected by the EPU conditions for the following reasons:

1. The Seismic loads are unaffected by the EPU;
2. No new pipe break locations or pipe whip and jet impingement targets are postulated as a result of the uprated condition;
3. Pipe whip and jet impingement loads do not increase for the EPU; and
4. SRV and LOCA dynamic loads used in the original design basis analyses are bounding for the EPU

10.2.3.1.1 Safety-Related SRV ~~and Power Operated Valves~~

The licensee performed the over-pressure protection analysis at the uprated power condition using 3 percent SRV setpoint tolerance. The staff finds the use of 3 percent tolerance to be

See comment next page

Note: This was a previously-approved license amendment and not the subject of staff review in this application.

acceptable for DAEC as the inservice surveillance testing of the plant's SRVs have not shown a significant propensity for the high setpoint drift greater than 3 percent. The analysis calculated a peak RPV steam pressure of 1313 psig at the bottom of the vessel. This peak pressure remains below the ASME allowable of 1375 psig (110% of design pressure) and that safety-related SRV operability is not affected by the proposed EPU. Furthermore, the maximum operation reactor dome pressure remains unchanged for the DAEC EPU. Consequently, the licensee concluded that the SRV setpoints and analytical limits are not affected by the proposed EPU, and that the SRV loads for the SRV discharge line piping will remain unchanged. The staff agrees with the licensee's conclusion that the SRVs and the SRV discharge piping will continue to maintain their structural integrity and to provide sufficient over-pressure protection to accommodate the proposed EPU.

10.2.3.1.2 ~~Safety-Related SRVs and~~ Safety-Related Power-Operated Valves

As discussed in its original request and response to staff questions, the licensee evaluated the effect of the EPU on the capability of power-operated valves to perform their safety functions at Duane Arnold. The licensee reviewed calculations and settings for the safety-related motor-operated valves (MOVs) within the scope of the programs established in response to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves." The review included potential effects of the EPU on the operating requirements for the valves and output of safety-related MOV motor actuators. In addition, the licensee is evaluating its air-operated valves (AOVs) as part of an industry-wide effort, and has confirmed that the EPU will not adversely affect the capability of AOVs at Duane Arnold to perform their safety functions. The licensee has also evaluated the potential pressure locking and thermal binding of its safety-related power-operated gate valves as a result of the proposed EPU. The licensee reported that the EPU condition did not impact its screening criteria established in response to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." The licensee evaluated the valves previously determined to be susceptible to pressure locking or thermal binding, and determined that the EPU would not adversely affect those valves. The staff finds the licensee's evaluation of the effect of the proposed EPU on the capability of safety-related power-operated valves at Duane Arnold to be acceptable. The licensee also indicated that the proposed EPU conditions are bounded by the current containment analysis and thus have no impact on the evaluation in response to GL 96-06 on potential overpressurization of isolated piping segments for DAEC.

Based on the information provided by the licensee, the staff concludes that the proposed EPU will not have an adverse effect on the performance of mechanical components of safety-related valves at DAEC.

10.3 Required Testing

10.3.1 Generic Test Guidelines for GE BWR Extended EPU

NEDC-32424P-A, Section 5.11.9, provides the general guidelines for EPU testing.

- A testing plan will be included in the uprate licensing application. It will include pre-operational tests for systems or components which have revised performance requirements. It will also contain a power increase test plan.

The licensee's test plan follows the guidelines of NEDC-32424P-A and the staff position regarding individual EPU amendment requests.

10.3.3 Systems/Components with Revised Performance Requirements

The guidelines in NEDC-32424P-A, Section 5.11.9, specify that pre-operational tests will be performed for systems or components which have revised performance requirements. These tests will occur during the ascension to EPU conditions. The performance tests and associated acceptance criteria are based on Duane Arnold's original startup test specifications and previous GE BWR EPU test programs. The licensee has identified performance tests for the following systems.

- Intermediate range neutron monitors assure SRMs and APRM overlap
- APRM calibration
- Pressure regulatory system setpoint steps, failures, incremental regulation
- Feedwater control system setpoint changes, incremental regulation
- Radiation measurements survey
- Feedwater system vibration
- Main steam system vibration

With regard to the steam pressure or recirculation flow testing, neither parameter has changed for the uprate program. Therefore, testing of system performance is not necessary.

The results from the uprate test program will be used to revise the operator training program to more accurately reflect the effects of the EPU.

10.3.4 Large Transient Tests

In a submittal dated June 11, 2001, Duane Arnold proposed to implement the EPU in two phases. EPU Phase I would result in 1790 MWt (112.4 percent of the ORTP and 107.9 percent of the current RTP). EPU Phase II would result in 1912 MWt (120 percent of the ORTP and 115.3 percent of the current RTP).

Section 5.11.9 of NEDC-32424P-A, "Power Uprate Testing," states that a MSIVC test, equivalent to that conducted in the initial startup testing, will be performed if the EPU is more than 10 percent above any previously recorded MSIVC transient data. NEDC-32424P-A also states that a generator load rejection test equivalent to that conducted in the initial startup testing will be performed if the EPU is more than 15 percent above any previously recorded generator load rejection transient data. DAEC experienced unplanned events at approximately 1658 MWt that provided the data to fulfill the requirements of Section L.2.4 of NEDC-32424P-A up to and including power levels of 1823.8 MWt for the MSIVC test and 1906.7 MWt for the generator load rejection test. Therefore, the large transient tests are not required for Phase I. However, consistent with NEDC-32424P-A these tests should be performed prior to Phase II.

The staff is currently evaluating the need for large transient tests for EPUs. The staff is conducting its evaluation under the Dresden 2 and 3 and Quad Cities 1 and 2 EPU reviews. The staff intends that its decision on the Dresden 2 and 3 and Quad Cities 1 and 2 applications would apply to DAEC. In addition, the DAEC licensee has committed to implement the Phase II test program consistent with the outcome of the staff's evaluation of the Dresden 2 and 3 and

Quad Cities 1 and 2 reviews. Because the staff has not completed its evaluation of this issue, this issue remains open for the Duane Arnold EPU.

The staff finds that there is reasonable assurance that the applicant's EPU testing program is consistent with the requirements of 10 CFR 50, Appendix B, and the NRC-approved topical report NEDC-3242P-A, Section 5.11.9 for a EPU of 1790 MWt . The staff is still evaluating the need for the large transient tests as described in NEDC-32424P-A. Therefore, the need for the licensee to conduct these tests prior to Phase II is considered an open item.

10.4 Individual Plant Examination

The license amendment application was submitted in accordance with ELTR-1 and ELTR-2. Consistent with ELTR-1, the licensee provided a plant-specific evaluation of the risks associated with their proposed EPU. The staff reviewed this risk information, using the guidelines delineated in RG 1.174. This evaluation included a review of the licensee's discussions on EPU impacts to core damage frequency (CDF) and large early release frequency (LERF) due to internal events, external events (i.e., fire and seismic), and shutdown operations. The evaluation also addressed the quality of the DAEC PRA, commensurate with its use in the licensee's and staff's decision-making processes.

10.4.1 Internal Events

Consistent with NEDC-3424P-A the licensee evaluated the changes due to EPU implementation for impact on the probabalistic risk assessment (PRA) models for internal events in the following key areas: initiating event frequency, component reliability, system success criteria, and operator response. Each of these areas is specifically addressed in the following subsections, followed by a description of the overall risk impacts from internal events for the EPU.

10.4.1.1 Initiating Event Frequency

For the original DAEC PRA, initiating event frequencies were developed for:

- Abnormal operating transients (AOTs), such as manual reactor scram, turbine trip, LOFW, loss of condenser vacuum, MSIVC, inadvertent opening of relief valves, and LOOP, including SBO.
- ATWSs, including MSIVC, LOFW, loss of condenser vacuum, and turbine trip.
- LOCAs, including small, medium, and large interfacing systems and LOCAs outside containment.
- Other special events, such as loss of 125 V dc power, loss of instrument air, loss of river water supply, and internal flooding.

These internal initiating events were qualitatively assessed by reviewing the underlying bases in relation to the frequency of occurrence of the elements and considering the potential impacts of the EPU.

action failed. The operator actions were further screened to exclude those errors that occurred before the initiating event (e.g., instrument calibration errors), those whose timing were not related to reactor power (e.g., operator actions based on battery capacity), and those in which the operators had separate hours to diagnose the problem and complete the tasks (so that the reductions in the times available to respond were negligibly small).

several

The screening identified the following five operator actions that were evaluated further for their impact on plant risk:

- Initiation of SLC for turbine trip and MSIVC ATWS events.
- Inhibiting the ADS for MSIVC ATWS events with high-pressure injection initially available.
- Initiation of reactor water level control in order to reduce power for MSIVC ATWS events.
- Initiation of SLC and power/level control for turbine trip ATWS events with bypass available.
- Depressurization of the vessel to allow low-pressure injection into the vessel following failure of the high-pressure injection systems for non-ATWS events with the reactor at high pressure.

These operator actions and the associated impacts due to the EPU are discussed below.

The first operator action addresses the reduction in time available for the operators to initiate SLC for turbine trip and MSIVC ATWS events. This scenario pertains to ATWS events with the main condenser not available as a heat sink. If injection by both SLC pumps is successful early on, then the need for reactor vessel emergency depressurization will be avoided, as the heat capacity temperature limit (HCTL) for the suppression pool will not be reached. If this operator action is not performed early in the scenario, the suppression pool's HCTL will be reached and emergency depressurization will be required. Still, if SLC injection from at least one SLC pump is initiated within a reasonable time, in conjunction with the initiation of suppression pool cooling (SPC), containment failure from high suppression pool temperature will be prevented. The two different time windows for this operator action determine the success criterion (i.e., number of trains) required later in the event analysis for SPC to avoid containment failure due to overheating. Early SLC injection leads to the need for only one train of SPC, while late SLC injection leads to the need for two trains of SPC. Due to the EPU, the early SLC initiation timing is reduced from 6 minutes to 4 minutes, while the late SLC initiation timing is reduced from 20 minutes to 14 minutes. Based on the reduction in available time, the HEP for early SLC initiation was increased from 1.1E-1 to 1.8E-1 and the HEP for late SLC initiation was increased from 7.5E-2 to 9.5E-2.

The second operator action addresses the reduction in time available for the operators to inhibit ADS in MSIVC ATWS events. This scenario pertains to ATWS events for which the main condenser is not available as a heat sink and the feedwater and condensate systems are not available for reactor inventory makeup. High-pressure injection, however, is initially available. The failure to inhibit ADS results in the automatic injection of a large quantity of water by the low-pressure ECCS. This failure would dilute the boron in the core, resulting in a recriticality.

The staff finds that the risk increases due to the reduced operator response times available under the EPU conditions are small and meet the guidelines of RG 1.174 for both internal and external events. Further, the staff finds that the licensee has a process for managing plant risk during shutdown operations and that the risk impact due to the EPU during these operations wil be negligible.

Based on the licensee's reported analyses and results, the staff concludes that the increases in CDF and LERF from internal, external, and shutdown events due to the proposed EPU are small and that the risk impacts are within the guidelines set forth in RG 1.174.

10.5 Human Factors

This evaluation is limited to the operator performance impacts expected from the increased maximum power level. It includes required changes to operator actions, human-system interface, procedures, and training resulting from the change in maximum power level. The evaluation is based on the licensee's responses to five broad questions regarding human performance.

The staff's guidance for this review includes Information Notice 97-78, "Crediting of Operator Actions In Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," and ANSI/ANS-58.8, "Time Response Design Criteria for Safety-Related Operator Actions," 1984.

10.5.1 Human Factors Review

(Reference 18)

The staff's evaluation of the licensee's responses to the five questions is provided below.

Question 1 - Describe how the proposed EPU will change the plant emergency and abnormal procedures.

The licensee responded that minor changes to curves and limits were made to the EOPs, but none of the changes have any impact on existing accident response strategies or require new operator actions. Similarly, the AOPs require minor changes, but no significant changes to operator actions have been identified. The staff finds these statements regarding changes to procedures to be satisfactory.

Question 2 (a) - Describe any new risk-important operator actions required as a result of the proposed EPU.

As stated above, no new risk-important operator actions have been identified.

Question 2 (b) - Describe changes to any current risk-important operator actions that will occur as a result of the EPU.

The licensee described six operator actions considered to be risk important in which the time available to perform the action will be decreased as a result of the EPU. Based on the criteria of ANSI/ANS-58.8, only one operator action, to initiate SLC in turbine trip and MSIVC ATWS events, appears to present an issue. If the action can be successfully completed within

6 minutes of the beginning of the event (4 minutes after EPU), the reactor vessel emergency depressurization will be avoided. The ANSI/ANS 58.8 criteria indicate that 9 minutes should be available for this action. However, if the licensee can satisfactorily justify a shorter time, the shorter time is acceptable.

In its letter dated July 11, 2001, the licensee provided a detailed description of the steps an operator would take to diagnose the need for SLC system injection and the execution of that action, including a discussion of the controls used to initiate SLC system injection and the ATWS EOP to be followed. The licensee further indicated that this scenario is a routine training exercise and that boron injection using SLC system before reaching the boron injection initiation temperature (BIIT) curve limit is a "critical task" in the training program. The licensee stated that training records from 1997 to present show that 58 evaluated scenarios involving ATWS events were conducted and that 100 percent of the crews successfully executed this task. It should be noted that while time to accomplish the injection task was not recorded in these simulations, all were successfully accomplished before reaching the BIIT curve limit. The licensee estimates the actual task to take between 10-15 seconds, well within the 4 minutes available with the EPU. The staff accepts this justification for the shorter available task time.

Question 2 (c) - Explain any changes in plant risk that result from changes in risk-important operator actions.

? (SPSB should respond to this question based on the NMC statement that failure of early SLCS injection is not a direct path to a core damage state.)

Question 3 (a) - Describe any changes to the operator interfaces for control room controls, displays, and alarms as a result of the proposed EPU. For example, what zone markings (e.g. normal, marginal, and out-of-tolerance ranges) on meters will change?

The licensee provided a comprehensive list of control room indications that will need to be rescaled or rebanded. In addition, the following items will be added or replaced: flow transmitters for hydrogen injection, feedwater pump suction indication, and reactor water cleanup flow transmitters.

Question 3 (b) - What setpoints will change?

The licensee stated that the following setpoints will change: APRM, main steamline high flow and high radiation isolations, and turbine first-stage pressure trip bypass.

Question 3 (c) - How will operators know of the change?

The licensee stated that all setpoint changes are made under the licensee's design control process, which requires that all design changes be evaluated by the training department for impact on both operator and craft training.

Question 3 (d) - Describe any controls, displays, and alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU.

15.0 REFERENCES

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2) General Electric Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," Licensing Topical Report NEDC-32424P-A (ELTR1), February 1999.

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3) General Electric Energy, "General Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," Licensing Topical Report NEDC-32523P-A (ELTR2), February 2000.

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5) General Electric, "General Electric Standard Application for Reactor Fuel," GESTAR II, NEDE-24011-P-A

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6) General Electric, "GE Fuel Bundle Design," NEDE-31152P, Volumes 1,2, and 3, December 1988.

7) General Electric Nuclear Energy, "BWR Owners Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," Licensing Topical Report NEDO-32465, May 1995.

8) General Electric Nuclear Energy, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," NEDO-32047-A, June 1995.

9) General Electric Nuclear Energy, "BWR Owners Group Long-term Stability Solution Licensing Methodology," NEDO-31960-A and Supplement 1, April 1996.

10) General Electric, "Assessment of BWR Mitigation of ATWS, Volume II" (NUREG-0460 Alternative No. 3), NEDE-24222, December 1979.

11) Nuclear Regulatory Commission, "Power Oscillation in Boiling Water Reactors (BWRs)," Bulletin Number 88-07, Supplement 1, December 1988.

12. Nuclear Regulatory Commission, "Standard Review Plan," NUREG-0800, April 1996

13) Letter from Stuart A. Richards, Nuclear Regulatory Commission, "Review of NEDC-32992P ODYSY Application for Stability Licensing Calculation," enclosing Safety Evaluation, April 20, 2001.

14) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-037-Alternative Source Term" (TAC # MB0347), dated March 23, 2001.

15) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Request for Operating License Change (TSCR-0400)-Revised Thermal-Hydraulic Analysis for the Spent Fuel Pool", (TAC # MB0596), dated April 9, 2001.

2 letters on this date: NG-01-0463 NG-01-0499

16) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated April 16, 2001.

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17) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated May 8, 2001.

18) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated May 10, 2001.

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19) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated May 11, 2001.

20) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated May 29, 2001.

21) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated June 5, 2001.

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24) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated June 21, 2001.

25) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated June 28, 2001.

26) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated July 11, 2001.

27) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated July 19, 2001.

28) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated July 25, 2001.

2 letters on this date: NG-01-0925 NG-01-0926

29) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated August 1, 2001.

30) Letter from Gary Van Middlesworth, Nuclear Management Company, to Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 - Extended Power Uprate" (TAC # MB0543), dated August 10, 2001.