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FACIL:50-331 Duane Arnold Energy Center, Iowa Electric Light & Pow 05000331
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MCGAUGHY,R.W. Iowa Electric Light & Power Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Application to amend License DPR-49,revising Tech Specs to allow power uprate to 1,658 mwt.Approval requested prior to startup for Cycle 8.Forwards NEDO-30603 & NEDC-30603-P. NEDC-30603-P withheld (ref 10CFR2.790).Fee paid.
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Iowa Electric Light and Power Company

August 17, 1984
NG-84-2518

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Power Uprate for the Duane Arnold Energy Center

Dear Mr. Denton:

Transmitted herewith, in accordance with the requirements of 10 CFR 50.59 and 10CFR50.90, is an application for amendment to Appendix A (Technical Specification) to Operating license DPR-49 for the Duane Arnold Energy Center.

The enclosed amendment request, RTS-165, has been reviewed by the Duane Arnold Energy Center Operations Committee and the Safety Committee. Per the revised fee schedule for license amendments (10 CFR 170), a check for $150 is enclosed. The balance of the application fee will be paid upon billing.

As implementation of this application involves changes to the facility which can only be performed during a refueling outage, we request that this application not be approved prior to shutdown for the Cycle 7/8 refueling outage. Also, once these modifications have been implemented, approval of this application will be necessary prior to the startup for Cycle 8.

A General Electric Company proprietary report: "Duane Arnold Energy Center Power Uprate Report" (NEDC-30603-P), May, 1984 is enclosed in support of this application. This report contains information which the General Electric Company customarily maintains in confidence and withholds from public disclosure and as such, has been handled and classified as proprietary to General Electric. The confidential information has been identified in the report by vertical bars within the margins. As indicated in the attached affidavit, we hereby request that NEDC-30603-P be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790(a)(4). A non-proprietary version of this report suitable for public disclosure has also been provided.

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In addition, a summary report outlining the results of our Balance-of-Plant review and a copy of our proposed test program to be conducted in support of this application are enclosed.

Three signed and 37 additional copies of this application are transmitted; five of the copies have the GE proprietary report NEDC-30603-P attached. These five copies are stamped in the upper right corner as "Company Proprietary". Pursuant to the requirements of 10 CFR 50.91, a copy of this application and analysis of no significant hazards considerations is being sent to our appointed state official. This application, consisting of the foregoing letter and enclosures, is true and accurate to the best of my knowledge and belief.

IOWA ELECTRIC LIGHT AND POWER COMPANY

BY Richard W. McGaughy

Richard W. McGaughy
Manager, Nuclear Division

Subscribed and sworn to Before Me on this 16th day of August 1984.

Kathleen M. Furman
Notary Public in and for the State of Iowa

RWM/RAB/rh*

Attachments:
1) Proposed Change RTS-165
2) Evaluation of Change per 10 CFR 50.92
3) General Electric Affidavit of Proprietary Information
4) NEDC-30603-P/NEDO-30603
5) Balance of Plant Summary Report
6) Proposed Test Program
7) Technical Specification Affected Pages

cc: R. Browning
L. Liu
S. Tuthill
M. Thadani
T. Houvenagle (ICC)
NRC Resident Office

Proposed Change RTS-165
to the
Duane Arnold Energy Center
Technical Specifications

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with attached, new pages. A List of the Affected Pages is given below.

The purpose of this change is to amend the Technical Specifications to allow the DAEC to operate at its' licensed, maximum, steady state core power level of 1658 MWt. In order to support this change, several areas of the Technical Specifications must be modified. These include redefinition of rated power, revised dome pressure-dependent setpoints, new APRM flow-biased setpoints, modified limits on the primary containment and updated ECCS limits.

The supporting analyses of design basis events and abnormal operating transients show that the original safety evaluation performed to support the original license application for 1658 MWt is still valid when compared to current, applicable acceptance criteria.

A brief description of the changes being made follows:

1) Update List of Figures to show new MAPLHGR curve for new fuel type for Cycle 8, correct the name on fuel type P8DPB289 and delete fuel types to be discharged at end of Cycle 7, (8D274L and H).

2) Revise definitions section to show that rated (100%) power is now 1658 MWt, that the pressure scram setpoint has been raised to 1055 psig, and to update the MSIV scram and isolation signals to agree with Amendment #89 (Low-Low Set Logic).

3) Revise the APRM flow-biased scram and rod block equations to allow operation above the rated (100%) load line at low power and flow conditions. New criteria for the allowable APRM gain adjustment for thermal peaking factor setdown is also included. Update figures 1.1-1, 2.1-1 to show the new APRM flow-biased scram and rod block lines. The Bases for this section are also updated to address these changes.

4) Revise the reactor pressure scram and relief valve setpoints to account for the 20 psi increase in dome pressure. Update the bases describing the vessel overpressure protection analysis.

5) Revise the bypass setting for turbine control valve and stop valve closure scram to correspond to 30% of the new rated power.

6) Revise the setpoint for ATWS recirculation pump trip on reactor high pressure to account for the increase in dome pressure.

7) Revise the surveillance requirement for the HPCI and RCIC systems to account for the increased dome pressure. Update the bases accordingly.

8) Update the allowable out-of-service time for one ADS valve to be consistent with the basis for the new LOCA analysis. Revise bases to indicate correct valve capacity and reference latest LOCA analysis.

9) Revise the bases in Section 3.6 to agree with the valve setpoints in Section 2.2 and to discuss the simmer margin analysis report, which is also added to the references.

10) Revise the temperature limit for conducting visual inspections of the suppression pool to be consistent with the modifications made for the Mark I improvement program. Update the bases accordingly.

11) Delete the section (4.7.A.7) dealing with the Initial Leakage Rate Tests conducted during the initial plant startup.

12) Revise the peak pressure for conducting the primary containment leakage rate testing in accordance with the new containment analysis. Update the bases per the new analysis and add the new analysis report to the references.

13) Update the fuel MAPLHGR curves per the results of the revised LOCA analysis. Update the references to reflect the latest LOCA analysis.

List of Pages Affected

vii	1.1-17	3.1-21	3.6-28	3.7-44
1.0-2	1.1-18	3.2-16	3.6-41	3.7-48a
1.0-3	1.1-21	3.2-17	3.7-1	3.7-49
1.0-5	1.1-23	3.2-23	3.7-3	3.12-1
1.1-1	1.1-25	3.5-6	3.7-4	3.12-11
1.1-2	1.2-1	3.5-8	3.7-6	3.12-16*
1.1-3	1.2-4	3.5-9	3.7-24	3.12-17
1.1-14	3.1-3	3.5-19	3.7-31	3.12-18
1.1-15	3.1-4	3.5-22	3.7-36	3.12-19
1.1-16	3.1-6	3.5-26	3.7-37	3.12-20

*These pages have been deleted.

EVALUATION OF CHANGE WITH RESPECT TO 10 CFR 50.92

In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based upon the following information:

The DAEC was designed and constructed to operate at a steady state core thermal power level of 1658 MWt. The original safety analysis report submitted as part of the operating license application justified operation at this power level. The NRC reviewed the application and in concluding that operation of the DAEC at 1658 MWt posed no significant hazards to the health and safety of the public, granted an Operating License for the DAEC at a steady state power level not to exceed 1658 MWt. However, the Technical Specifications issued as Appendix A to the Operating License limited reactor operation to 1593 MWt. This application, when approved, will amend the Technical Specifications to allow operation of the DAEC at the originally licensed power level of 1658 MWt. The supporting analysis, using currently approved analytical models and acceptance criteria, shows that the original safety analysis report is still valid and that operation of the DAEC at a steady state core power level of 1658 MWt poses no significant hazards to the health and safety of the public.

In the April 6, 1983 Federal Register, the NRC published examples of amendments that are not likely to involve a significant hazards concern. Example number (iv) of that list states:

"A relief granted upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation was not yet demonstrated. This assumes that the operating restriction and the criteria to be applied to a request for relief have been established in a prior review and that it is justified in a satisfactory way that the criteria have been met."

The Safety Evaluation Report (SER) issued with the Operating License for the DAEC specifies the acceptance criteria for requesting relief from the restriction limiting core power to 1593 MWt. This application, including a description of a test program called for in the SER, meets the criteria specified in the previous review. Therefore, this example is judged to apply to this application.

The following are more detailed 50.92 evaluations for each of the various areas of the Technical Specifications which require amending to support this application:

A. Redefine Rated Power

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The DAEC was originally designed, analyzed and licensed for a steady state core power of 1658 MWt. Thus the probability of an accident is not increased above those in the UFSAR by redefining rated (100%) power as 1658 MWt. The supporting analyses of the design basis events and abnormal operating transients was conducted at the new rated conditions. The results of which show that the consequences of an accident are not increased above those previously analyzed in the UFSAR by redefining rated (100%) power as 1658 MWt.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

The steady state power level is not the initiating event of any accident. Therefore, redefining rated power does not create the possibility of a different type of accident than those previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

The supporting analyses of design basis events and abnormal operating transients were conducted at the new steady state power including a 2% uncertainty factor in power level in accordance with Reg. Guide 1.49. The results of these analyses show that the margin of safety is not reduced by increasing the steady state core power level, ie. by redefining the rated power level.

In the April 6, 1983 Federal Register, the NRC published examples of amendments that are not likely to involve a significant hazards concern. Example number (vi) of that list states:

> "A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

The supporting analysis shows that increasing the steady state core power meets all the applicable acceptance criteria. In several instances the results are improved over those in the original Safety Analysis Report due to refinements in calculational models and design methods developed and approved since the original application. Thus, the above example is judged to apply.

approved since the original application. Thus, the above example is judged to apply.

B. Reactor Pressure - Dependent Setpoints

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The reactor dome pressure will increase 20 psi (due to the higher steam flow) at the new rated conditions. Therefore, all dome pressure-dependent setpoints will have to be increased by at least this amount to give the same operating margins. Since the primary system is designed for a much higher pressure, the probability of an accident is not increased above that in the UFSAR. The supporting analyses of design basis events and abnormal operating transients show that the consequences of an accident are not increased above those previously analyzed in the UFSAR by increasing the dome pressure 20 psi and adjusting the pressure-dependent setpoints accordingly.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

As the primary system was originally designed and analyzed for operation at the new dome pressure, the possibility of an accident of a different type than any previously evaluated is not created by increasing the dome pressure and adjusting the pressure-dependent setpoints.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

The supporting analyses of design basis events and abnormal operating transients were conducted with the revised instrument setpoints based on the 20 psi increase in reactor dome pressure. The results of these analyses show that the margin of safety is not reduced by increasing the dome pressure and raising the pressure-dependent instrument setpoints an equivalent amount.

The ATWS Recirculation Pump Trip Setpoint was increased 20 psi from 1120 to 1140 psig. The generic ATWS analysis for BWR/4's (NEDE-24222) used a setpoint of 1150 psig. Thus, revising the DAEC setpoint has no impact upon the generic analysis.

In the April 6, 1983 Federal Register, the NRC published examples of amendments that are not likely to involve a significant hazards concern. Example number (vi) of that list states:

"A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

As the primary coolant system was designed to operate at the new system pressure and the supporting analyses show that all the acceptance criteria, primarily the ASME Pressure Vessel Code overpressure protection criteria, are met, the above example is judged to apply.

C. APRM Setpoints and Gain Adjustment Factor

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Adjusting the flow-biased equations for APRM scram and APRM & RBM rod blocks will allow operation above the 100% load line at less than rated power/flow conditions. Operation above the 100% load line is achieved by withdrawal of control rods at low power/flow conditions using pre-established withdrawal sequences. The only accidents initiated by withdrawal of control rods are the Rod Withdrawal Error and Control Rod Drop Accidents, which require a withdrawal of a rod out-of-sequence and a decoupling of the control blade from its' drive, respectively. Both of these events are independent of the withdrawal sequence used or final rod pattern chosen. Thus, the probability of these events is not increased from that analyzed in the UFSAR by operating above the 100% load line. The supporting analyses show that operation above the 100% load line is bounded by the analyses conducted at the 100% power and 100% flow conditions, except for the Feedwater Controller Failure transient. However, the Feedwater Controller Failure is not the most-limiting transient for determining the operating limit MCPR's. Thus, the consequences of an accident are not increased from those previously analyzed in the UFSAR.

Adjusting the APRM flow-biased equations downward when the Maximum Fraction of Limiting Power Density (MFLPD) exceeds the Fraction of Rated Power (FRP) by increasing the APRM gain has been shown to be equivalent to setting down the scram and rod block setpoints by modifying the circuitry (the original method). This original method was cumbersome, inefficient and complicated. Since adjusting the equations by increasing the APRM gain is simpler, the probability of operator error and hence the probability of an accident is actually reduced from that previously analyzed in the UFSAR. Adjusting the gain upward on the APRM's causes the instruments to read higher values than are actually experienced and limits have been placed on the allowable gain adjustment, such that the APRM's can never indicate a reading greater than 100% power. Since all accidents and transients are assumed to be initiated at 102% power the consequences of an accident will not be increased above these previously analyzed in the UFSAR.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

As adjusting the APRM flow-biased scram and rod blocks to allow operation above the 100% load line directly impacts only the Rod Withdrawal Error and Control Rod Drop Accidents, both of which are analyzed in the UFSAR, the possibility of a different type of accident is not created.

Adjusting the APRM gain upward is equivalent to setting down the APRM flow-biased scram and rod blocks to account for high thermal peaking factors. This gain adjustment causes the APRM's to read higher than the actual values experienced. This is conservative as it causes the APRM readings to be closer to the trip setpoints (this is the intent of the thermal peaking factor setdown). Since APRM gain adjustment is a normal part of plant operation as described in the UFSAR, the possibility of an accident of a different type from those previously analyzed is not created.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

The supporting analyses of design basis events and abnormal operating transients were conducted above the 100% load line (100% Power, 87% Flow). In all cases, except the Feedwater Controller Failure (FWCF) transient, the results were bounded by those analyzed at the 100% power, 100% flow condition and thus the margin of safety for these events are not reduced above those previously analyzed. While the results of the FWCF transient at the (100,87) point were slightly worse than those at rated conditions, the margin of safety is not degraded as this a non-limiting event and is not used for determining the operating limit MCPR's, which define the margin of safety.

In the April 6, 1983 Federal Register, the NRC published examples of amendments that are not likely to involve a significant hazards concern. Example number (vi) of that list states:

> "A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

Operation above the 100% load line has been analyzed and found to meet all the acceptance criteria for accidents and abnormal operating transients. Therefore, the above example is judged to apply.

Also, example number (vii) states:

> "A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more stringent surveillance requirement."

Currently, there are no restrictions in the Technical Specifications limiting the amount of allowable APRM gain adjustment. Therefore, this example is judged to apply to this change.

D. <u>Containment Limits</u>

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The peak pressure at which the primary containment will be leak-rate tested is based upon the expected maximum pressure to be experienced after a Loss-of-Coolant Accident (LOCA). As the new peak pressure is calculated to be lower than that in the previous analysis in the UFSAR, the probability of occurrence or magnitude of the consequences of an accident is not increased.

The current limit on the maximum allowable suppression pool temperature after an S/RV discharge without performing an inspection of the torus is based upon the limits for stable steam condensation from a ramshead discharge device. As part of the NRC-approved Mark I containment upgrades, the DAEC replaced these ramsheads with T-quenchers devices, which have a higher temperature limit for stable steam condensation. Thus, the increased temperature limit for suppression pool temperature is being added to be consistent with the new hardware and therefore does not increased the probability or consequences of an accident previously analyzed in the UFSAR.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

The primary containment is leak rate tested at the maximum expected pressure following a LOCA to ensure that containment integrity will be maintained after such an accident. Thus reducing the test pressure based upon a new analysis of containment response following a LOCA will not introduct the possibility of an accident of a different kind than previously evaluated.

The limit on suppression pool temperature following an S/RV discharge is for requiring a visual inspection <u>after</u> the event has taken place. Thus, raising the limit for determining when the inspection is required, will not introduce the possibility of an accident of a different type than previously analyzed.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

As the new peak pressure for leak-rate testing the primary containment, based upon the new LOCA analysis, is lower than the previous value, the margin of safety is not reduced.

The visual inspection is required whenever conditions in the suppression pool are reached which could generate large hydrodynamic loads that could cause structural damage to the torus. The new temperature limit is based upon the stable condensation limits of the T-quencher devices installed as part of the Mark I program and is consistent with the basis for the old temperature limit associated with the ramshead devices. Thus, the margin of safety as defined in the basis of any technical specification is not reduced.

In the April 6, 1983 Federal Register, the NRC published examples of amendments that are not likely to involve a significant hazards concern. Example number (vi) of that list states:

"A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in

some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

The lower peak pressure for leak rate testing the primary containment is based upon the results of an analysis of containment response following a LOCA. This analysis used the improved calculational models developed as part of the Mark I containment program. These improved models have been approved for this application by the NRC staff. Therefore, this example is judged to apply.

Also, example number (vii) states:

"A change to make a license conform to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations."

As part of the Mark I containment program, the DAEC installed T-quencher devices on the safety/relief valve (S/RV) discharge lines to insure stable steam condensation following an S/RV discharge. The NRC, thru NUREG-0661, placed limits on local suppression pool temperatures for plants with T-quencher devices. This change to the Technical Specification is to revise the current limit on suppression pool temperature following an S/RV discharge to conform to the new limits. Therefore, this example is judged to apply to this change.

E. <u>ECCS Limits</u>

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

In issuing License Amendment #16, the NRC in their Safety Evaluation Report stated that "Since the licensees did not provide a LOCA analysis with one ADS valve out-of-service for small line breaks, a Technical Specification has been revised so as not to allow continuous operation for more than seven days with any ADS valve out-of-service."

The supporting analysis of small break LOCA's (SBLOCA) was conducted with one ADS valve out-of-service and thus the Tech. Specs. are being revised to allow operation of up to thirty days (the previous limit) with one ADS valve out-of-service. Increasing the allowable out-of-service time of one ADS valve does not increase the probability of an accident (a SBLOCA). The supporting analysis shows that the consequences of an accident (SBLOCA) are not increased over those previously analyzed.

The revised MAPLHGR limits are based upon the supporting analysis and ensure that the consequences of an accident are not increased above those previously analyzed in the UFSAR. In addition revised MAPLHGR limits on the fuel do not increase the probability of an accident previously analyzed (LOCA) as they are only operational restrictions to limit the consequences of the accident.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Failure of a single ADS valve has previously been analyzed, thus increasing the allowable out-of-service time for a single ADS valve will not introduce the possibility of an accident of a different type than previously analyzed.

(3) Does the proposed amendment involves a significant reduction in a margin of safety?

The function of the ADS valves are to provide relief capability in the event of a SBLOCA with a concurrent failure of the HPCI system, so that the low pressure ECCS may be initiated to limit the peak fuel temperature to less than 2200°F. The supporting analysis shows that enough relief capacity exists such that the 2200°F limit can be maintained with the additional failure of a single ADS valve. Therefore, the allowable out-of-service time may be relaxed to thirty days without degrading the margin of safety.

The new MAPLHGR limits are based upon the supporting LOCA analysis, which shows that as long as the fuel is operated at less than or equal to these thermal limits the peak fuel temperature following a LOCA will be less than the 2200°F limit. Therefore, these new MAPLHGR limits ensure that the margin of safety is not reduced.

In the April 6, 1983 Federal Register, the NRC published examples of amendments that are not likely to involve a significant hazards concern. Example number (iv) of that list states:

> "A relief granted upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation was not yet demonstrated. This assumes that the operating restriction and the criteria to be applied to a request for relief have been established in a prior review and that it is justified in a satisfactory way that the criteria have been met."

In the Safety Evaluation Report for DAEC License Amendment #16 the NRC imposed the restriction on the allowable out-of-service time for a single ADS valve, as the supporting analysis to that amendment was not analyzed with this additional single failure. The supporting analysis to this application was analyzed with the additional single failure of one ADS valve and demonstrates that the acceptance criteria (10 CFR 50.46) previously established were met. Therefore, this example is judged to apply to this change.

Also, example number (vi) states:

> "A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

The new MAPLHGR limits on the fuel are established using currently approved licensing models (10 CFR 50 App. K) and ensure that the acceptance criteria (10 CFR 50.46) are met at the new operating conditions. Therefore, the above example is judged to be applicable to this change.

GENERAL ELECTRIC COMPANY

AFFIDAVIT

I, Glenn G. Sherwood, being duly sworn, depose and state as follows:

1. I am Manager, Safety and Licensing Operation, General Electric Company, and have been delegated the function of reviewing the information described in paragraph 2 which is sought to be withheld and have been authorized to apply for its withholding.

2. Duane Arnold Energy Center Power Uprate, April 1984 (NEDC-30603-P).

3. In designating material as proprietary, General Electric utilizes the definition of proprietary information and trade secrets set forth in the American Law Institute's Restatement Of Torts, Section 757. This definition provides:

   "A trade secret may consist of any formula, pattern, device or compilation of information which is used in one's business and which gives him an opportunity to obtain an advantage over competitors who do not know or use it.... A substantial element of secrecy must exist, so that, except by the use of improper means, there would be difficulty in acquiring information.... Some factors to be considered in determining whether given information is one's trade secret are: (1) the extent to which the information is known outside of his business; (2) the extent to which it is known by employees and others involved in his business; (3) the extent of measures taken by him to guard the secrecy of the information; (4) the value of the information to him and to his competitors; (5) the amount of effort or money expended by him in developing the information; (6) the ease or difficulty with which the information could be properly acquired or duplicated by others."

4. Some examples of categories of information which fit into the definition of proprietary information are:

   a. Information that discloses a process, method or apparatus where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

   b. Information consisting of supporting data and analyses, including test data, relative to a process, method or apparatus, the application of which provide a competitive economic advantage, e.g., by optimization or improved marketability;

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c. Information which if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality or licensing of a similar product;

d. Information which reveals cost or price information, production capacities, budget levels or commercial strategies of General Electric, its customers or suppliers;

e. Information which reveals aspects of past, present or future General Electric customer-funded development plans and programs of potential commercial value to General Electric;

f. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection;

g. Information which General Electric must treat as proprietary according to agreements with other parties.

5. In addition to proprietary treatment given to material meeting the standards enumerated above, General Electric customarily maintains in confidence preliminary and draft material which has not been subject to complete proprietary, technical and editorial review. This practice is based on the fact that draft documents often do not appropriately reflect all aspects of a problem, may contain tentative conclusions and may contain errors that can be corrected during normal review and approval procedures. Also, until the final document is completed it may not be possible to make any definitive determination as to its proprietary nature. General Electric is not generally willing to release such a document to the general public in such a preliminary form. Such documents are, however, on occasion furnished to the NRC staff on a confidential basis because it is General Electric's belief that it is in the public interest for the staff to be promptly furnished with significant or potentially significant information. Furnishing the document on a confidential basis pending completion of General Electric's internal review permits early acquaintance of the staff with the information while protecting General Electric's potential proprietary position and permitting General Electric to insure the public documents are technically accurate and correct.

6. Initial approval of proprietary treatment of a document is made by the Subsection Manager of the originating component, the man most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within the Company is limited on a "need to know" basis and such documents at all times are clearly identified as proprietary.

7. The procedure for approval of external release of such a document is reviewed by the Section Manager, Project Manager, Principal Scientist or other equivalent authority, by the Section Manager of the cognizant Marketing function (or his delegate) and by the Legal Operation for technical content, competitive effect and determination of the

accuracy of the proprietary designation in accordance with the standards enumerated above. Disclosures outside General Electric are generally limited to regulatory bodies, customers and potential customers and their agents, suppliers and licensees only in accordance with appropriate regulatory provisions or proprietary agreements.

8. The document mentioned in paragraph 2 above has been evaluated in accordance with the above criteria and procedures and has been found to contain information which is proprietary and which is customarily held in confidence by General Electric.

9. The information contained herein is the result of extensive analyses performed at considerable cost to the General Electric Company. The development and verification of these methods, as well as their application and execution cost in excess of $100,000.

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:

Glenn G. Sherwood, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 1 day of JUNE, 1984.

Glenn G. Sherwood
General Electric Company

Subscribed and sworn before me this 1st day of June 1984.

Ruthe M. Kinnamon
NOTARY PUBLIC, STATE OF CALIFORNIA

OFFICIAL SEAL
RUTHE M KINNAMON
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My comm. expires APR 26, 1985
175 Curtner Avenue, San Jose, CA 95125

JLR:cal/K053027*-3
5/30/84

DAEC POWER UPRATE PROGRAM
BALANCE-OF-PLANT EVALUATIONS

An evaluation of DAEC Balance-of-Plant (BOP) systems was performed to support the DAEC power uprate program.(1) The BOP system study consisted of a review of existing DAEC documentation, including the UFSAR, system descriptions, equipment specifications, calculations, and vendor drawings, to determine whether BOP systems were originally designed based on the uprated power level. The study concluded that BOP systems that are normally designed based on plant operation at a particular power level were designed based on uprated power (i.e. 1658 MWt). For those systems, such as HVAC systems, which are designed based on process conditions or other parameters not directly related to power level, system design was based on parameters that envelop power uprate conditions. No modifications, or changes to controls or set points, to support operation at uprated power were identified.

In addition, an evaluation was performed to determine the potential impact of uprated power on bounding Design-Basis-Accident (DBA) dose assessments, control room habitability, and the DAEC Environmental Qualification (EQ) program. The results of these evaluations are as follows:

- Radiological effects of bounding DBAs at 102% of uprated power are within the guidelines of 10 CFR 100.
- Control room doses resulting from postulated accidents at 102% of uprated power are within the guidelines of 10 CFR 50, Appendix A, Criterion 19.
- Environmental qualification of equipment contained in the DAEC EQ program is not affected by environmental conditions associated with uprated power.

(1) Attachment 1 to letter L. Lucas to R.E. Lessly, "Power Uprate BOP Study Report," June 18, 1984.

POWER UPRATE TEST PROGRAM DESCRIPTION

The following is a description of the test program to be conducted at the DAEC following Cycle 7 - 8 refueling. The test program includes those tests which will be performed while bringing reactor power from zero to 1593 MWt and those additional tests which will be performed while increasing power level from 1593 to 1658 MWt. Using the data from the tests conducted at 1593 MWt, pre-test predictions will be made for the expected results at 1658 MWt. These pre-test predictions will be compared with the actual results and any large discrepancies will be resolved prior to steady state operation at 1658 MWt.

The tests, inspections, and verifications described below are standard surveillance tests performed after each refueling outage. Some of the procedures to be performed after core alterations are completed before criticality. Others are performed during startup at various power levels. A number of the tests will be performed just prior to exceeding 1593 MWt and repeated at 1658 MWt.

The tests are generally presented in the order in which they would be performed beginning with completion of core alterations.

When "prior to exceeding 1593 MWt" is referred to, the test will be conducted between 1500 and 1593 MWt.

A. Zero to 1593 MWt testing

RSCS and RWM checks - 43B003*

The purpose of this test is to demonstrate operability of the RSCS and RWM systems. This test will be performed prior to rod withdrawal for the initial criticality of the cycle 8 core.

SRM trip functional test and calibration - 42C005

The purpose of this test is to demonstrate the operability of the Source Range Monitor upscale rod block functions. This test is performed within one week prior to startup.

IRM trip functional test and calibration - 41A004

The purpose of this test is to demonstrate the operability of the inoperative, downscale, upscale alarm, and upscale trip functions on the Intermediate Range Monitoring system and the rod block logic system. This test is performed weekly while the mode switch is in startup or refuel mode and within one week before startup.

*These numbers refer to the Surveillance Test Procedure Number.

APRM high flux (15% scram) instrument functional test and calibration - 41A017

The purpose of this test is to demonstrate operability of the Average Power Range Monitoring system high flux, inoperative, and rod block functions when the mode switch is not in the run mode. This test is performed weekly during refueling or startup and before each startup.

SRM/IRM detector not in startup position functional test - 42C004

The purpose of this test is to demonstrate operability of the source and intermediate range detector not fully inserted rod block functions of the Neutron Monitoring System. This test is performed within one week of startup.

Reactor High Pressure recirculation pump trip instrument functional test and calibration - 42G001

The purpose of this test is to demonstrate operability of the reactor high pressure recirculation pump trip instrument channels of the reactor recirculation system. This test is performed once per operating cycle.

Shutdown margin test - 43A001

The purpose of this test is to demonstrate that the reactor can be made subcritical at any time during the fuel cycle, with a margin of 0.38% K/k, with the highest worth operable control rod fully withdrawn. The test is performed during the first pull to critical following core alterations. The analytically determined highest worth rod is pulled out fully. The shutdown margin is verified. The reactor is then pulled to critical.

Control rod friction testing and insert/withdraw timing - RE #15

The purpose of this procedure is to measure control rod insert and withdraw times to determine if excessive friction exists in the control rods. This test is performed after core alterations and at 0 psig. This test is not required by the plant technical specifications. This test is performed as part of the preventive maintenance program.

Heatup and cooldown rate - 46A003

The purpose of this test is to permanently record vessel shell, bottom drain, recirculation loop, and bottom head temperatures during heatup and cooldown.

Scram insertion time test - 43C001

The purpose of this test is to demonstrate operability of the scram insertion function of the Reactor Protection System for all control rods. This test is performed prior to attaining 40% power and at greater than 950 psig saturated.

MSIV trip and closure time check - 47D004

The purpose of this test is to demonstrate that closure time of the MSIVs is between 3 and 5 seconds for those valves which were manipulated during the refuel outage. This test is performed at reduced power (75%) to accommodate the reduction in steam flow caused by valve closure.

B. 1593 to 1658 MWt testing

APRM gain adjustment - 42F007

The purpose of this calibration is to correlate the gain of the APRM channels with the actual thermal power of the reactor core. This calibration is performed prior to reaching 20% power and daily thereafter. Just prior to exceeding 1593 MWt, the APRM gain adjustment will be performed and will be continued on a daily basis thereafter. Also, this adjustment will be made just prior to reaching 1658 MWt.

LPRM instrument calibration - 41A015

The purpose of this test is to calibrate the Local Power Range Monitors by reference to the Traversing Incore Probe System. This test is performed every 1000 effective full power hours (894 MWD/ST). At approximately 40% power and prior to exceeding 1593 MWT, an LPRM instrument calibration will be performed.

Axial Power Distribution

The purpose of this test is to compare the axial power distribution predicted by the General Electric 3D BWR core simulator code and the onsite predictor code to the actual P1 axial power distribution on the plant process computer. This test will be performed at 1593 MWt and repeated at 1658 MWt. This test is not required by the plant Technical Specifications but is performed to verify the analytical tools available at the DAEC.

Reactivity analomies check - 43D001

The purpose of this test is to compare the critical rod configuration at specific power operating conditions with the configuration expected based upon appropriately corrected past data. This test is performed every full power month. This test will be performed prior to exceeding 1593 MWt and after reaching 1658 MWt.

Reactor coolant chloride ion and conductivity analysis - 46B003

The purpose of this test is to sample the reactor coolant for:

1. A chloride ion check at least every four hours during startup and at steaming rates below 100,000 lbs/hr if the conductivity is above 0.5 mho/cm or if it increases at a rate of 0.2 mho/cm/hr or more;

2. A chloride ion check at least daily during startup and steaming below 100,000 lbs/hr under normal conductivity levels; and

3. A conductivity and chloride ion check at least every four days when fuel is in the reactor vessel.

This test will be performed prior to exceeding 1593 MWt and at 1658 MWt. The test will be performed according to the routine surveillance requirements in between these two power levels.

Reactor coolant gamma and iodine activity - 46B001

The purpose of this test is to sample the reactor coolant system for gross gamma activity including:

1. Taking a reactor sample at least every 96 hours and detemining gross gamma activity of the filtrate. I-131 and I-133 concentration are determined weekly.

2. Taking a sample prior to startup and at four hour intervals during the startup when equilibrium iodine is equal to or greater than 0.012 Ci/gm of dose equivilant I-131.

3. Taking a sample and analyzing for I-131 equivalent dose when the equilibrium iodine value is equal to or greater than 0.012 Ci/gm of dose equivalent I-131 and the gaseous waste monitor prior to holdup indicates an increase of greater than 50% in the steady state fission gas release.

4. Taking a sample and analyzing at four hour intervals for gross iodine when equilibrium iodine is equal to or greater than 0.012 Ci/gm of dose equivilant I-131 and following a power change of greater than or equal to 15%.

5. Providing a means to determine if additional sampling is required.

Appropriate startup sections of this test will be performed prior to exceeding 1593 MWt and after reaching 1658 MWt. At power levels in between, appropriate surveillances for startup will be performed.