UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555

June 9, 2004

Mr. Tony Browning
3313 DAEC Road
PSC - Regulatory Affairs
Palo, IA 52324

Mr. Browning,

After your conversation with Mr. Caruso of the ACRS staff, he asked that I send you a copy of the material you requested in your e-mail dated May 28, 2004. I am enclosing the referenced memo and attachments with the exception of the information classified as **"GE Proprietary."**

Hope this information satisfies your request.

Sincerely,

Michele S. Kelton

Michele Kelton
Technical Information Assistant

Enclosure:
Memo dated October 3, 2001, from P. Boehnert, Senior Staff Engineer, to ACRS Members, Subject: ACRS Review of Duane Arnold Core Power Uprate Request - Additional Information

cc:
R. Caruso, ACRS
M. Snodderly, ACRS
Deann Raleigh, SCIENTECH, Inc.

Duane Arnold
251

October 3, 2001

MEMORANDUM TO: ACRS Members

FROM: P. Boehnert, Senior Staff Engineer

SUBJECT: ACRS REVIEW OF DUANE ARNOLD CORE POWER UPRATE REQUEST - ADDITIONAL INFORMATION

The Committee will review the license amendment request of the Nuclear Management Company (NMC), for a core power uprate for the Duane Arnold Energy Center (DAEC) on October 4, 2001. The Thermal-Hydraulic Phenomena Subcommittee reviewed this matter during its September 26-27, 2001 meeting. The following additional information is provided in support of the Committee's review:

- Presentation Schedule for Committee Discussion of the DAEC core power uprate.

- Excerpt from the Transcript of the September 26-27, 2001 T/H Phenomena Subcommittee Meeting wherein Dr. Powers summarizes the Subcommittee's concerns with the staff's review process as evidenced by the meeting discussions.

- Report of ACRS Consultant V. Schrock on the September 26-27, 2001 T/H Phenomena Subcommittee Meeting. Professor Schrock raises some issues that also were of concern to the Subcommittee pursuant to NRR's review process (see above).

- Response from the Nuclear Management Corporation (NMC - DAEC Licensee), to a question posed by Dr. Wallis during the September 26-27 subcommittee meeting with regard to the procedure used to calculate material stresses.

- NRC Technical Evaluation Report on the results of an audit of the NMC calculation of energy response for the containment, given a large-break LOCA. The audit found that the licensee's calculational methodology acceptable. This issue will be discussed by the staff during its presentation.

- Memorandum from J. Zwolinski, NRR, providing responses to questions from the T/H Phenomena Subcommittee regarding the DAEC core power uprate (**contains proprietary information**).

Attachments: As Stated

cc w/o attach (via E-mail):
 J. Larkins
 S. Bahadur
 H. Larson
 R. Savio
 S. Duraiswamy
 ACRS Technical Staff & Fellows

[redacted]

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
486th MEETING
DUANE ARNOLD ENERGY CENTER CORE POWER UPRATE REQUEST
OCTOBER 4, 2001
ROCKVILLE, MARYLAND

PRESENTATION SCHEDULE

Contact: P. Boehnert (301/415-8065)
("pab2@nrc.gov")

TOPIC	PRESENTER	TIME
I. Subcommittee Report	D. Powers, ACRS Cognizant Member	8:35 a.m.
II. Duane Arnold Power Uprate		
A. NMC Presentations		
• Introduction	R. McGee, NMC	8:45 a.m.
• Plant Changes to Accommodate Power Uprate		
• Compliance with Regulatory Requirements		
• Training		
• Thermal-Hydraulic Analyses - ATWS - Stability/Instability - Containment - NPSH - ECCS	T. Browning, NMC	
• Materials		
• PRA Analyses		
B. NRR Presentation		9:30 a.m.
1. Introduction	B. Mozafari	
2. NRR SER Review		
- Reactor & Fuel Performance/ GE Audits	R. Caruso	
- Cumulative Usage Factors	K. Manoly	
- Evaluation of Containment Response	R. Lobel	
- Concluding Remarks	B. Mozafari	
III. Committee Caucus/Recess		10:15 a.m.

innovative things there, too. Now, I would like to come to the staff presentation at this meeting.

And I will begin again with my suggestion to the committee, and see if they will overrule me, just as efficaciously as they did with respect to the applicant.

It seems to me that opposing sets of questions for the subcommittee meeting, in the interest of efficiency, we may have sandbagged the staff a little bit. And that we need to give them more freedom to design their presentation.

And I would encourage them to design their presentation to dissuade the committee from writing a letter that begins, "With the ACRS unable to ascertain if the staff has done an adequate review of the Duane Arnold application for a power uprate. Our examination of the SER suggests the staff has asked perceptive, probing questions. Documentation of the resolution of these questions in the SER is quite limited has become the familiar pattern for SERs."

"Our discussions with the staff did not produce satisfactory amplification of the SER. Too often the staff appears to have accepted a methodology that has been proven in the past without showing that it has also done an adequate investigation into the application of the approved methods."

"After oral discussion with the staff, it is not apparent that the staff is adequately familiar with either the methods or the specific application."

I think that I would like the staff to make a presentation that forecloses writing that kind of a letter.

CHAIRMAN WALLIS: In 45 minutes.

DR. POWERS: In 45 minutes.

CHAIRMAN WALLIS: With questions.

DR. POWERS: With questions. I think the areas that the subcommittee has pursued in here give you some guidance to what we are looking for when we say have you done an adequate application or investigation on how it was applied to the specific issue here.

I think we are in general familiar with those approaches that the staff has accepted in the past, and it is really how they were applied that is at issue here.

And as I said, when I read the SER, I found -- my general impression in reading the SER were the questions that the staff was asking were the right questions. In fact, they were very good.

It's that their final resolution doesn't come through as clear and clarifying. I am giving you my personal viewpoint, and I will turn to the rest of the committee and see what they would like to hear from the staff.

DR. KRESS: Personally, I will ~~bite~~ BVY off ~~from~~ ON what you said. That would have been my recommendation.

DR. POWERS: Professor Wallis, have you any guidance that would like to give the staff on their presentation?

CHAIRMAN WALLIS: Well, I think you have given them a challenge. I'm just wondering how they will respond to it. I guess I will just have to wait and see.

DR. POWERS: I remain confident that they can, because again I looked at the SER, and I looked at the kinds of questions that were being asked, and addressed, and I thought that they were perceptive and challenging questions.

CHAIRMAN WALLIS: The only thing that I worry about is the committee getting into some of the morass that we got into; is that when we start probing the rationale for the decisions, we have difficulty getting answers to the questions posed. I don't want that to happen with the full committee. The answer should be crisp and to the point and reassuring.

DR. POWERS: Professor Schrock, can you give us some help here?

DR. SCHROCK: Probably not. I have been concerned for a long time about this issue of the falling back on the fact that analyses are done in accordance with previous approvals, and frequently that gets in the way of communicating an understanding of what is done and how it is applied in the present situation. I think you have said that very well.

And I am glad to hear that challenge thrown up to the staff. I think that is something that needs to change and it needs very badly to change.

So apart from my strong feeling on that, I don't think I can give you a lot of guidance on how you are going to cope with your problem of getting all this information exchanged in this short period of time.

DR. POWERS: And Dr. Ford.

DR. FORD: I have four specific questions that you can pass on to the staff.

DR. POWERS: Oh.

DR. FORD: You are giving them a challenge, and I am giving them four specific questions to help them meet the challenge.

DR. POWERS: Very good. Do you want to share them with us?

DR. FORD: Well, we have already gone through it in the other meeting. It is the CDF situation and FIC, and FAC, and the corrosion/ erosion cracking. I can give them to you. I have gotten them written out.

DR. POWERS: Okay.

MR. SHUAIBI: Dr. Powers, can I ask a question?

DR. POWERS: Certainly.

MR. SHUAIBI: This is Mohammed Shuaibi of the staff again. Is it your perception that the entire safety evaluation is this way, or is it just inadequate in certain areas?

DR. POWERS: I did not in the course of the presentation find an area that we asked questions in that I thought was handled in a way that was reassuring. Well, I take that back. I found the answers to the NPSH margin questions by the section head were answered promptly and explicitly.

MS. KAVANAUGH: Thank you.

DR. POWERS: Now, the criterion question that Dr. Wallis asked still is more nebulous, but I don't know that you are responsible for that in this application. Okay. Any other comments that the members would like to make?

Have we given you -- I'm sure that we haven't given you enough, but would you like to hear me talk anymore?

MS. MOZAFARI: No, I think we have an idea. We will go back and revisit our conclusions, and our evaluations to make sure that we have been clear enough about the basis for the evaluations.

DR. POWERS: Feel free to interact with Mr. Boehnert, who will be in a position to pass on any clarifications that you might need.

MS. MOZAFARI: Okay.

DR. POWERS: With that, I will turn the meeting back to Professor Wallis.

CHAIRMAN WALLIS: I would like to thank the representatives from Duane Arnold and GE, and the staff, and my colleagues for their contributions to this meeting, and I will adjourn the meeting.

(Whereupon, the opening meeting was recessed at 12:20 p.m.)

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September 29, 2001

To: Dr. Graham B. Wallis
Chairman, Subcommittee on Thermal Hydraulic Phenomena
Advisory Committee on Reactor Safeguards
Via: Paul Boehnert
From: Virgil E. Schrock, Consultant

Subject:: T/H Subcommittee Meeting September 26 & 27, 2001
Extended Power Uprate for Duane Arnold

The owner of Duane Arnold, NMC, has requested an Extended Power Uprate that will bring the licensed power level to 20% above the original licensed power. The power had been previously uprated by 5%. The additional increase will be accomplished following the so-called constant pressure uprate strategy. The power increase is obtained by flattening the radial power profile and by adding bundles. The existing maximum linear power remains unchanged. Steam flowrate is increased in peripheral bundles where the power is increased. Core flowrate is said to remain the same. The power increase impacts the system behavior in operational transients and accident response (RCS and Containment). It also reduces the time available for operators to interpret plant condition and take action in critical events. Presentations made by the NMC personnel and by GE personnel made the case that the requested uprate has no impact on safety margin and that operation at EPU is acceptable. The arguments seem reasonable for the most part but the amount of supporting data and many details of the analysis methods seemed incomplete.

LOCA analysis of the RCS was done using SAFER/GESTER, a "realistic" code and a strategy developed in the 1980s in response to SECY 83-472 which was the first step in the transition to the rule change to provide best estimate with evaluated uncertainty as an optional licensing basis. The details of the method were covered only superficially. An example is the decay power. The 1979 ANS/ANSI Standard provides methods for evaluating the <u>local</u> decay power (density) and its uncertainty as a function of the <u>local</u> operating power history. The emphasis is on <u>local</u> because decay power depends on the fissile nuclide source of the fission products. Fissioning in 235-U, 238-U (fast fission) and 239 Pu are accounted for separately by the 1979 ANS/ANSI Standard (241-Pu, for which there was no experimental data, is to be lumped with 235-U). Since the local concentrations of these nuclides are time dependent in reactor cores, the relationship between local decay power and local power history is a local one and varies throughout the reactor. Both the decay power and the uncertainty in our knowledge of it depend upon the relative fission rates in the fissile nuclides present and they are dependent upon time and position in the reactor. LOCA codes lack the detail in the core representation to analyze the decay power source as a space-time dependent input to the transient analysis of the thermal hydraulics. This level of detail, which is necessary to achieve an accuracy on the order of few percent, was ignored in earlier assessments such as GE's May-Witt, which preceded a regulatory process for licensing, and the 1971/1973 draft ANS Standards. May-Witt was one of several "estimates" of decay power that were used in the development of the 1971 draft ANS Standard. The accuracy (estimated) in the 1971/73 ANS draft Standard was 20% for the first 1000 seconds after shutdown and was based simply on the range of the available data and models about the selected mean. The draft ANS Standard gave decay power after shutdown following "infinite operation at constant power" but provided a method for assessing the decay power following finite reactor operation at constant power (finite operating time giving lower decay power). This provision was not included in the Appendix K requirement for decay heat of "ANS + 20%" (with no time limit on the 20%), so the law added conservatism to the conservative Standard. The 1979 Standard provided a sound basis for decay power evaluation with a statistically based uncertainty of a few percent (one sigma). The complications involved in the correct interpretation and application of the Standard data have prevented its full utilization in industrial and regulatory calculations. GE's approach is probably the best in the industry.

In the initial use of the 1979 Standard in SAFER/GESTER, GE used a generic decay power curve (core wide power after shutdown) derived from the 1979 Standard and an "uncertainty" whose basis was less clear, i.e., it was not a direct statistical combination of uncertainties given in the Standard but some sort of measure of the spread in a large number of core wide decay power evaluations that consider the space-time history of local fractions of fissioning in the contributing nuclides At the Meeting last November I asked GE if they had to redo this evaluation for the new fuels GE12 and GE14. The answer was not a crisp yes, but a "probably". At this meeting we were not shown the new generic curve and its evaluated uncertainty. I am not saying that I believe that this aspect of the GE LOCA analysis is not adequate but rather that the amount of new information that I received is inadequate for me to be sure. SAFER/GESTER became one of the approved codes/methods at a time when NRC gave less detailed scrutiny to code contents and bases than in recent years. It was not given sufficient scrutiny in the present review. In my view there are other ways in which the case presented for a 20% EPU lacks detail and sufficient quantitative results to fully support the overall conclusions.

In the TRACG review for AOO, a single curve representing the 1979 ANS/ANSI Standard was compared with the May-Witt formula for decay power. There is no such unique curve representing the 1979 Standard, for the reasons explained above. This is an example of a frequently used misconception about decay power. Decay power treatment should be more fully explained when the LOCA version of TRACG is reviewed next year.

In discussing the results of peak clad temperature one of the speakers said that some plants have LBPCT that push the 2200 deg. limit . Duane Arnold has a considerably lower value. I suggest that ACRS seek amplification of this issue. The licensing basis for Duane Arnold (and some other plants) is SAFER/GESTER. However for some plants it is still SAFE, an older version of the code, and I understand that it is for these plants that LBPCT is close to the 2200 deg. limit. What are the differences between SAFE and SAFER that account for the different LBPCTs?

The staff had a large number of questions posed in RAIs and the owners and GE replied to all of them. The staff found the replies acceptable and in the draft SER indicate agreement with the overall conclusions. The Staff's reasons for judging some things acceptable are less than clear in many cases. The NRC policy that previously approved methods may not be challenged has resulted in the SERs focusing too much on showing that approved methods have been used as opposed to a serious technical examination of the claims made to support a new licensing decision. Furthermore, many of the details in approved methods for Appendix K licensing are now little understood by people now involved, and some scrutiny now appears necessary. Some of the regulatory feeling of security seems to stem from the conventional wisdom that the decay power in Appendix K had enough conservatism to cover other features that were less well understood. In fact some requirements may not have been always conservative. The process needs improving.

The ACRS often has to provide comments on a review based on draft SERs. Final SERs (if they exist) are not likely to be reviewed again. Follow on requirements, stated or implied, are sometimes imposed in SERs without a clear mechanism for follow up. In the discussion on the SER for TRACG AOO version, NRR said they would revise the SER. Has that been done? Will it be? There appear to be loose ends in the Duane Arnold EPU SER too.

A more minor comment about the SER on Duane Arnold is that it is in need of editorial work. For example, it describes the uprate as 120% of the original power. Reference is made to the "1985 EPU" where it probably meant the first power uprate in 1995. There is enough of this sort of thing to make it clear that the document had inadequate editorial review.

The bottom line is that I believe that the power uprate for Duane Arnold is probably safe but I don't think the technical information provided is sufficiently complete to clearly prove this conclusion. I agree that GE should be commended for their openness in dealing with questions about their codes. I also think that NRR has done quite well, given the pressures of time. However the problems described above remain.

ACRS Thermal-Hydraulic Subcommittee Question

Why did the RPV closure flange stress increase ~10% when there is no pressure increase for this EPU?

DAEC Response

This question is based on information provided in our letter of April 16, 2001 (NG-01-0463), which provided our response to a Staff RAI. In this response, the EPU value for the RPV closure flange stress is 12.5% higher than the pre-EPU value. However, the EPU value is not an actual stress.

The methodology used in the evaluation of stresses for EPU effects starts with a very conservative screening process. The screening process determines scaling factors based on changes in loads. For example, pressure and temperature are assumed to have linear effects on stress, and all of the transient events are reviewed for changes due to EPU. For a particular plant component, the largest scaling factor calculated is then applied to the pre-EPU stress intensity and the result is compared to the applicable ASME Code allowable. This very conservatively applies the largest scaling factor to all of the loads, while maintaining the current acceptance criteria. Thus, the EPU value is not an actual calculated stress, but a demonstration of acceptability at EPU conditions. If the ASME Code allowable is exceeded during the screening process, then more rigorous methods would be employed to evaluate the actual stresses at EPU conditions.

For the RPV closure flange, the limiting scaling factor (SF) is a ratio of delta T (Temperature) for the transient before and after uprate (from the RPV thermal cycle diagram together with that specified in the EPU design specification.) For the Main Closure Flange, the limiting transient events of record are Turbine Trip and Over-Pressure with Delayed Scram (i.e., a full MSIV closure with direct scram failure.) Both transients have an 8-degree delta temperature in pre-EPU conditions. Both transients have a 9-degree delta temperature for the EPU condition. Therefore, the temperature SF is 9/8 = 1.125. Since this SF is applied to the bounding load or stress, there is a 12.5% increase in stress pre-EPU to EPU. However, the differential temperature magnitude and the 1°F increase are very little actual stress. Therefore, applying the 1.125 scaling factor to the pre-EPU stress intensity is extremely conservative, yet the ASME Code allowable is still satisfied.

Supporting Details

Scaling Technique

GE has developed a technique to conservatively scale the original stress report stresses to account for changes in the original pressures, temperatures, and nozzle flows as a result of EPU.

Many pressure vessel calculations select the three stress directions of the orthogonal coordinate system such that the shear stress components are zero; the normal stress components are the principle stresses. With this orientation, the pressure stresses are directly proportional to the increase in coolant pressure, and the magnitude of the principal stress resulting from thermal cycling is proportional to the temperature change during a thermal transient. When there are no changes in mechanical loads as a result of the EPU, the new magnitude of the principle stress is:

$$\sigma_{new} = \sigma_p * (P_{new}/P_{old}) + \sigma_t * (\Delta t_{new}/\Delta t_{old}) + \sigma_m$$

Where, σ_p = Original pressure stress
σ_t = Original thermal stress
σ_m = Original mechanical stress

Or:

$$\sigma_{new} = \sigma_p * SCF_p + \sigma_t * SCF_t + \sigma_m$$

Where, SCF_p = Pressure stress scaling factor

SCF_t = Thermal stress scaling factor

Most stress reports, including the DAEC's, do not separately report the individual pressure, thermal, and mechanical stresses; therefore, it is not practical to calculate the scaled pressure or scaled thermal stresses separately. A conservative scaling technique, using the larger of the pressure and temperature scaling factors, is used to scale the entire stress magnitude. If a calculated SCF is less than unity, a SCF = 1.0 is used instead. This method is a conservative alternative to scaling the individual stress components because:

1. The largest scaling factor is used for both the pressure and temperature SCF.

2. The mechanical stresses are increased by the SCF even though the design mechanical loads did not increase.

3. Conditions which generate a stress reduction (a SCF less than 1.0) are ignored.

These inherent conservatisms will offset any uncertainties in the calculation of the delta temperatures from the transient analyses.

Example: Main Closure Flanges:

For this case, the maximum applicable scaling factor was the thermal stress scaling factor (SCF_t), which was derived from the thermal cycle diagram, and conservatively used for both SCF_p and SCF_t, in the analysis of the closure flanges. All stress intensities and allowables used in the analysis were taken from the original DAEC stress report. Using the scaling methodology, the EPU Primary Plus Secondary Stress Intensity (P + Q) and bearing stresses are determined below:

Main Closure Flange - Head:

$$(P + Q)_{new} = (P + Q)_{old} * SCF_t$$

$$77{,}364 \text{ psi} = 68{,}768 \text{ psi} * 1.125$$

$$77{,}364 \text{ psi} < 3S_m = 80{,}100 \text{ psi}$$

If this conservative method had resulted in the Code allowables being exceeded, then a more detailed (realistic) calculational approach would be applied. The original stress calculation would be duplicated to generate the individual stress components for pressure, temperature, and mechanical loadings. The approach would be to apply the thermal scaling factor only to the thermal stress and apply a separate pressure scaling factor to the pressure stress and none to the mechanical stress to demonstrate margin to the Code allowables.

ISL-NSAD-NRC-01-001

TECHNICAL EVALUATION REPORT

Duane Arnold Energy Center
Extended Power Uprate Containment Analysis
Audit Calculation

B. J. Gitnick
Information Systems Laboratory, Inc.

Prepared for

U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

July 2001

Contract No. NRC-03-95-026
TAC No. MB0543

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ABSTRACT

An audit calculation has been performed for the Duane Arnold Energy Center (DAEC) long-term response of the Duane Arnold containment to a double-ended guillotine break of a recirculation line. These calculations used mass and energy input values and plant specific parameters furnished by the licensee and obtained from the Duane Arnold Energy Center Updated Safety Analysis Report. Although CONTAIN is a best-estimate code, these audit calculations used conservative assumptions. The results of these 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 the two calculations agree fairly well. The long-term CONTAIN calculation results in suppression pool conditions that are approximately 0.01 MPa (1.2 psia) and 2 °K (4 °F) higher than the GE results. Sensitivity studies have shown that these small differences can be explained by small changes in any one of several input values.

TABLE OF CONTENTS

Glossary

BWR	Boiling Water Reactor
CS	Containment Spray
DBA	Design Basis Analysis
DAEC	Duane Arnold Energy Center
ECCS	Emergency Core Cooling System
EPU	Extended Power Uprate
GE	General Electric Company
LOCA	Loss-of-Coolant Accident
MSLB	Main Steam Line Break
NMC	Nuclear Management Company, LLC
NPSH	Net Positive Suction Head
SRP	Standard Review Plan
SP	Suppression Pool
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report

TECHNICAL EVALUATION REPORT

Duane Arnold Energy Center
Extended Power Uprate Containment Analysis
Audit Calculations

1.0 INTRODUCTION

The Duane Arnold Energy Center (DAEC) is a General Electric (GE) designed Boiling Water Reactor (BWR). On November 16, 2000 Nuclear Management Company, LLC (NMC) requested a technical specification change to raise the rated thermal power of DAEC to 1912 MWt from its current limit of 1658 MWt (a 15.3% increase). This change is referred to as an Extended Power Uprate (EPU), since the rated power for DAEC had been uprated once before. Combined, these two power uprates raise the rated thermal power 120% from the original licensed value of 1593 MWt. In support of this technical specification change, NMC submitted containment analyses performed by GE at the EPU condition. The GE calculations showed a large margin to the containment design criteria related to drywell and wetwell peak pressures and temperatures. However, credit for containment pressure was required to ensure that the available Net Positive Suction Head (NPSH) for the ECCS pumps remains within limits. Because the requested power increase was much larger than those previously approved by the NRC, and because the GE codes have not been explicitly reviewed for this type of containment response application, audit calculations were performed.

These calculations used mass and energy input values provided by the licensee and plant specific parameters furnished by the licensee and obtained from the Duane Arnold Energy Center Updated Safety Analysis Report. Although CONTAIN is a best estimate code, these audit calculations used conservative assumptions similar to those used by the licensee.

The DAEC EPU does not change the reactor liquid mass inventory or operating pressure. However, other operating conditions will be changed which may significantly affect the containment response. The increased steam and feedwater flow at the higher power level will increase the core inlet subcooling about 7%. This greater subcooling increases the mass and energy release during the blowdown phase of LOCA events, increasing the short-term peak pressure and temperature. Because GE conservatively assumes that the initial reactor coolant is completed saturated for the long-term analysis, the EPU impacts the long-term containment response (suppression pool temperature and pressure) mainly due to the increase in decay heat following a LOCA or MSLB. These are also the dominant parameters for the available NPSH margin calculations. Therefore, the key results examined by the audit calculations were the drywell/wetwell pressure and temperature response for the short term analysis, and the suppression pool temperature for the long term calculation.

2.0 BACKGROUND

GE uses two different calculation procedures to predict containment response:

- Short- term pressure-temperature response to a DBA-LOCA is evaluated with the GE M3CPT code.

- Long-term analysis of the suppression pool temperature and available NPSH is performed by GE using the SHEX code.

GE uses different sets of non-mechanistic assumptions to conservatively model each case. For example, GE selects initial conditions which maximize the short-term blowdown mass and energy release for the case used to predict the peak containment pressure. However, GE maximizes the total energy release to containment for the long-term case by assuming the vessel inventory is saturated at the start of the LOCA event, even though this results in less mass released during the blowdown. Heat storage in structures is also conservatively neglected. (GE does include heat structures when performing NPSH calculations, as well as other assumptions that artificially minimize the pressure).

3.0 CONTAIN MODEL

The audit calculation was performed using the CONTAIN2.0 code [Ref. 1]. A previous Sandia study [Ref. 2] recommended a modeling procedure for CONTAIN code audit calculations. This procedure was followed where possible. As stated above, it was decided to make assumptions similar to the licensee's for these audit calculations. Therefore, actual GE break flow mass and energy release rates. and GE RHR and LPCI flows and energies were used as boundary conditions, instead of approximating mass and energy data from UFSAR information.

Most of the data needed to construct the CONTAIN model (mass and energy release rates, dimensions, loss coefficients, initial conditions, and modeling assumptions) were provided by GE in response to staff requests [Refs. 3-7]. As some of this information is considered GE proprietary, the construction of the CONTAIN input file was documented in a separate (proprietary) report [Ref. 8]. Geometric data not affected by the EPU (elevations and flow path lengths) were obtained from data and drawings provided in the DAEC UFSAR Section 6.2 [Ref. 9].

4.0 CONTAIN RESULTS

Figures 1-4 show CONTAIN results for a DBA LOCA double-ended guillotine break of a recirculation line. Figures 1 and 2 shows the long-term CONTAIN model results compared to GE's SHEX code. SHEX predicts slightly higher pressures and temperatures during the initial period, however, excellent agreement for suppression pool temperature and pressure were obtained after 40 minutes. The long-term CONTAIN calculation results in suppression pool conditions that are approximately 0.01 MPa (1.2 Psia) and 2 °K (4 °F) higher than the GE results. One exception was an apparently non-physical rise of 4-5 °K (8 °F) in the wetwell atmosphere temperature from 28800 seconds (8 hours) until 42000 seconds (11.7 hours) in the SHEX results. When questioned by the staff, GE responded that the anomalous behavior was this result of an inappropriate convergence parameter in the SHEX code. GE further stated that the improper setting of this convergence parameter did not impact the suppression pool temperature, which was the critical result for this calculation.

Sensitivity studies have also shown that the initial discrepancy in pressures and temperatures for the long-term case can mainly be ascribed to differences in modeling of mist/droplet retention in the drywell atmosphere. A portion of the small disagreement in long-term suppression pool pressures and temperatures can also be attributed to interpolation inaccuracies resulting from input of mass and energy data in CONTAIN as a linear multi-segment table.

CONTAIN has a mechanistic aerosol model that calculates droplet condensation and atmospheric liquid retention. Most containment codes (including SHEX) use simpler nonmechanistic models for condensation of atmospheric liquid. The action of containment sprays, which quickly condense atmospheric droplets, may mask this issue. As these cases conservatively assumed the sprays were unavailable, the CONTAIN mechanistic model relies on overflow out the break to act as a catalyst for droplet condensation. However, the magnitude and size distribution of the droplets resulting from post-reflood overflow out the break has considerable modeling uncertainty. The results shown in Figures 1 and 2 use the CONTAIN "dropout" option which removes atmospheric water by dropping out liquid every time step. Sensitivity studies of droplet related parameters have shown that the CONTAIN results for the initial 2400 second (40 minute) period can vary over a range which spans the GE results.

Figures 3 and 4 show short-term (0-30 seconds) results calculated with the CONTAIN short-term model compared to GE's M3CPT code. The short-term containment model differed from the long-term model in three ways:

- Different mass and energy release data were used.

- The initial vent submergence was changed to reflect the highest allowable suppression pool level (the long term model assumes that the suppression pool is at the lowest allowable level).

- Droplet parameters were changed to retain atmospheric liquid indefinitely (effectively the opposite of the "dropout" option).

The remaining model parameters were unchanged.

5.0 CONCLUSIONS

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. Tables 1 and 2 compare the results obtained for the key parameters. Even though the CONTAIN and GE methodologies had some differences, generally very good agreement was obtained.

6.0 REFERENCES

1) K. K. Murata, et. al., "Code Manual for CONTAIN2.0: A Computer Code for Nuclear Reactor Containment Analysis," NUREG/CR-6533, SAND97-1735, Sandia National Laboratory, December 1997.

2) K. K. Murata, et. al., "CONTAIN Code Qualification Report/User Guide for Auditing Design Basis BWR calculations," Draft Final Report, Sandia National Laboratory, October 1999.

3) E. D. Schrull, "Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate", NEDC-32980P (Proprietary), General Electric Co., November 2000.

4) G. Van Middlesworth, DAEC Site Vice-President, Attachment I of Letter to NRR, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 – Extended Power Uprate. (TAC # MB0543)", Control Number NG-01-0660, (Proprietary) Duane Arnold Energy Center, May 11, 2001.

5) G. Van Middlesworth, DAEC Site Vice-President, Attachment I of Letter to NRR, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 – Extended Power Uprate. (TAC # MB0543)", Control Number NG-01-0721, (Proprietary) Duane Arnold Energy Center, May 29, 2001.

6) G. Van Middlesworth, DAEC Site Vice-President, Attachment I of Letter to NRR, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 – Extended Power Uprate. (TAC # MB0543)", Control Number NG-01-0738, (Proprietary) Duane Arnold Energy Center, June 5, 2001.

7) G. Van Middlesworth, DAEC Site Vice-President, Attachment I of Letter to NRR, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042 – Extended Power Uprate. (TAC # MB0543)", Control Number NG-01-0805, (Proprietary) Duane Arnold Energy Center, June 28, 2001.

8) B. J. Gitnick,CONTAIN2.0 Model for Duane Arnold Uprate Audit Calculations, ISL Calculation File 5411-009-01 (Proprietary), Information Systems Laboratory, June 29, 2001.

9) Updated FSAR for Duane Arnold Energy Center, Docket #50-331.

Table 1. Short-Term Case
Comparison of Key Parameters

Parameter	M3CPT	CONTAIN
Peak Drywell Pressure	0.412 MPa [59.7 psia]	0.417 MPa [60.5 psia]
Peak Drywell Temperature	414.3 °K [286.1 °F]	415.7 °K [288.6 °F]
Time of Peak	4.31 seconds	4.30 seconds

Table 2. Long-Term Case
Comparison of Key Parameters

Parameter	SHEX	CONTAIN
Long Term Peak Suppression Pool Pressure	0.243 MPa [35.2 psia]	0.251 MPa [36.4 psia]
Long Term Peak Suppression Pool Temperature	375.0 °K [215.3 °F]	377.0 °K [218.9 °F]
Time of Peak	29555 seconds [8.2 hours]	32100 seconds [8.9 hours]

Figure 1. DAEC Uprate Long Term Case
Compartment Pressures vs Time

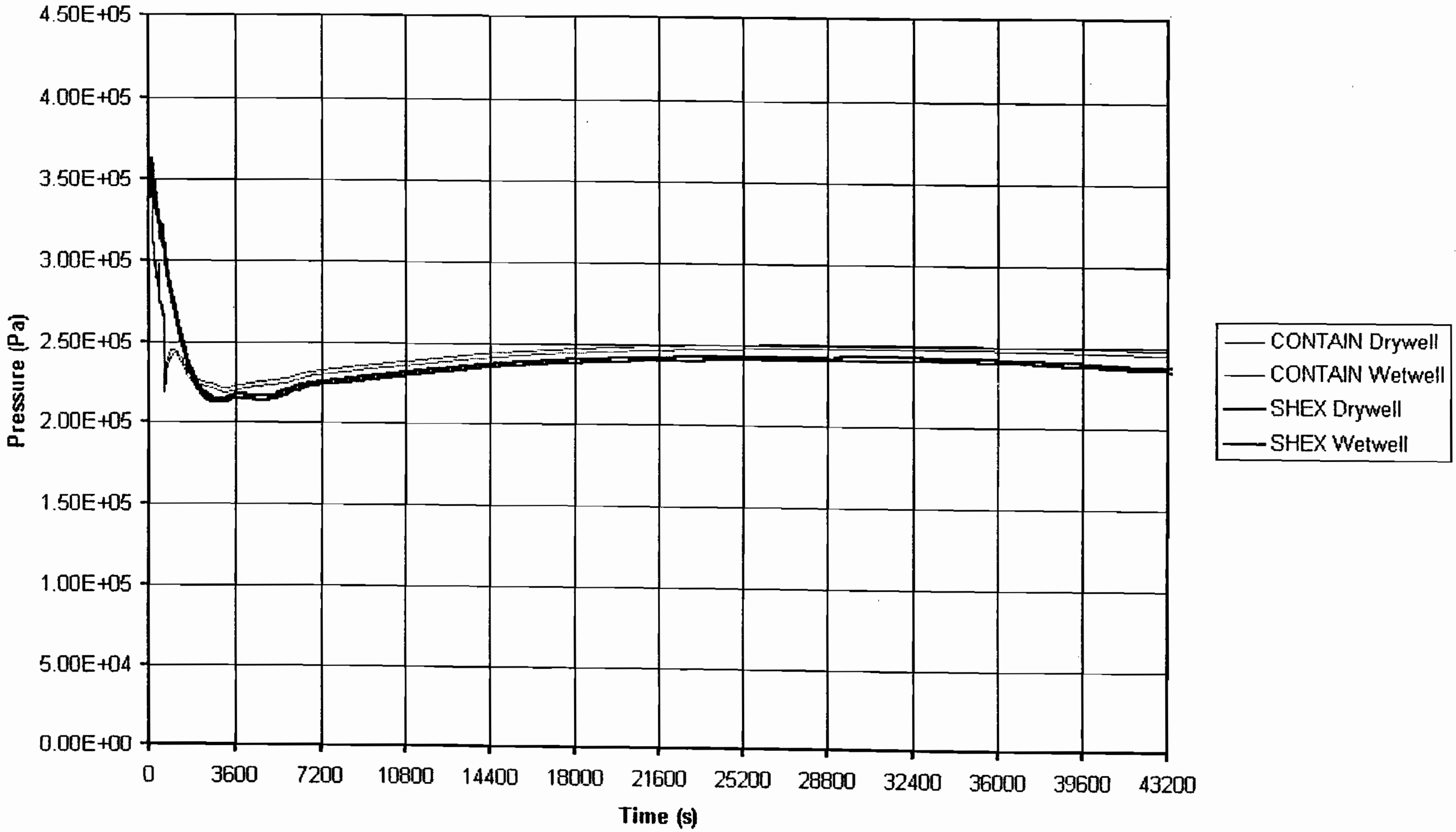


Figure 2. DAEC Uprate Long-Term Case
Temperatures vs Time

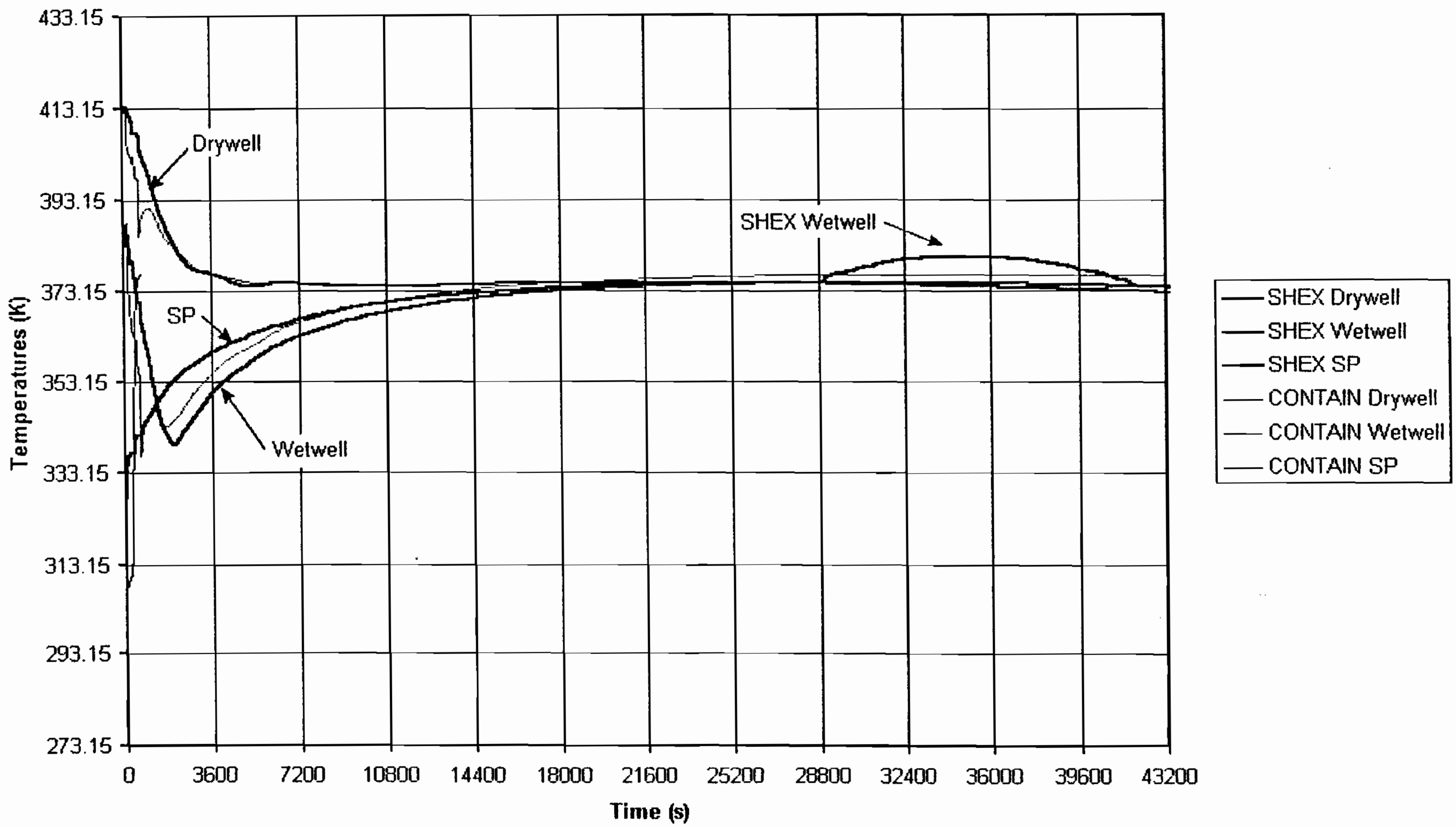


**Figure 3. DAEC Uprate Short Term Results
Compartment Pressures vs Time**

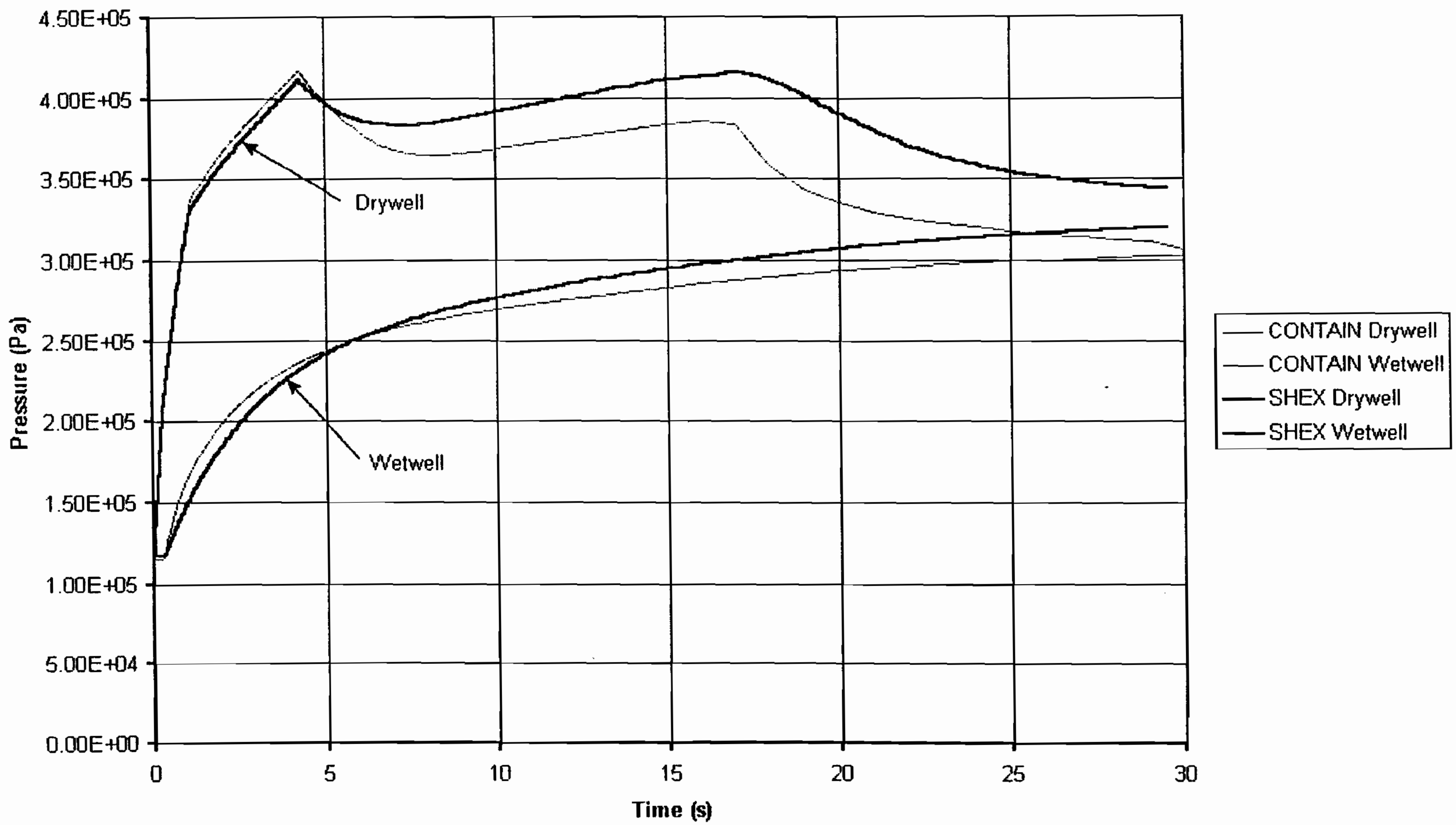
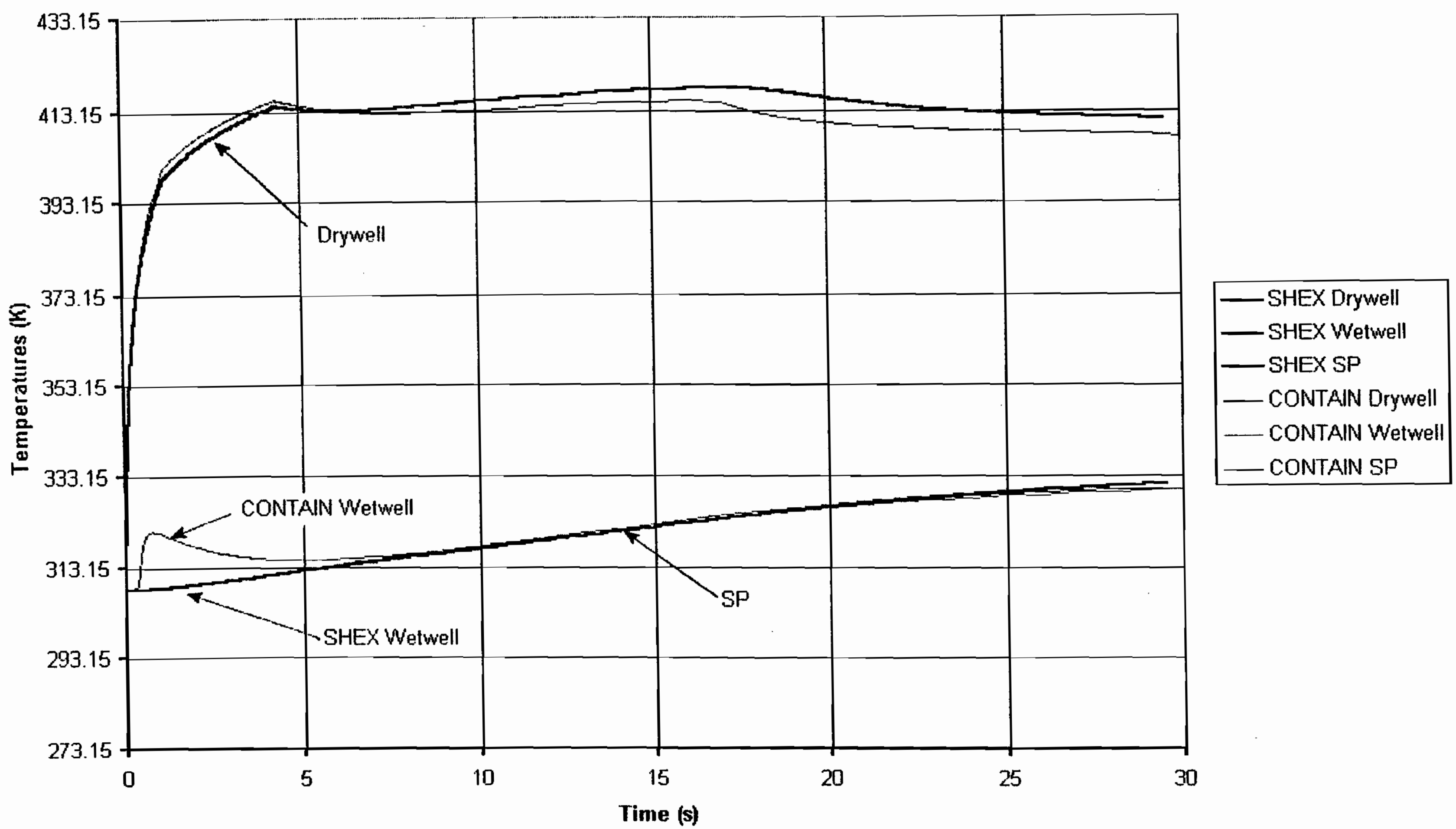


**Figure 4. DAEC Uprate Short Term Results
Temperatures vs Time**



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards and
Advisory Committee on Nuclear Waste

FROM: John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: RESPONSES TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) SUBCOMMITTEE QUESTIONS REGARDING DUANE ARNOLD ENERGY CENTER EXTENDED POWER UPRATE

Attached are responses to requests for additional information regarding the Duane Arnold Energy Center (DAEC) extended power uprate (EPU) of 15.3 percent. The attached responses by the Mechanical and Civil Engineering Branch (EMEB) and by the Materials and Chemical Engineering Branch (EMCB) are provided as a result of the September ACRS Thermal-Hydraulic Phenomena Subcommittee meeting and in support of the October ACRS Full Committee meeting.

The attached responses contain proprietary information. Therefore, I request that they not be released to the public.

Attachments: 1. EMCB Responses for DAEC EPU
2. EMEB Responses for DAEC EPU

cc: Paul Boehnert

CONTACT: Brenda Mozafari
415-2020

EMCB Responses to ACRS Subcommittee Requests on DAEC EPU

Request: Explain how CHECWORKS is applied for flow-accelerated corrosion (FAC) of carbon steel components.

Response:

The CHECWORKS is a predictive code. It is used to predict material loss due to FAC:

(a) for the components which were not included in the inspection program (no measurement of wall thickness was made), and

(b) for future projected wall thickness for all the components included in the program.

The prediction of the wall thickness is made in three steps:

Step 1 Initial wall thickness is determined by the code using plant model and specific data for each individual component. The plant model is developed using: diagrammatic representations of the plant, power level, quality of steam, chemistry, and available temperature and velocity values. The code will calculate these parameters from energy and mass balances for the portions of the plant where this information is not available. Flow velocity is determined by a special module in the code called Network Flow Analysis.

The specific data for the components include the geometry factor (affecting degree of turbulence) and the material factor (depending on the composition material - materials containing 1.25 percent or more of Cr are immune to FAC).

Step 2 In order to obtain final predictions, the initial wall thickness, determined by the code, has to be corrected using Line Correction Factor (LCF). The LCF represents a correction factor, based on measured values of wall thickness collected during the several previous outages. The LCF is determined for each individual line. Where the line is defined as a portion of the system having all components exposed to the water having the same temperature and chemistry (but not necessarily velocity).

The LCF is determined in the following way:

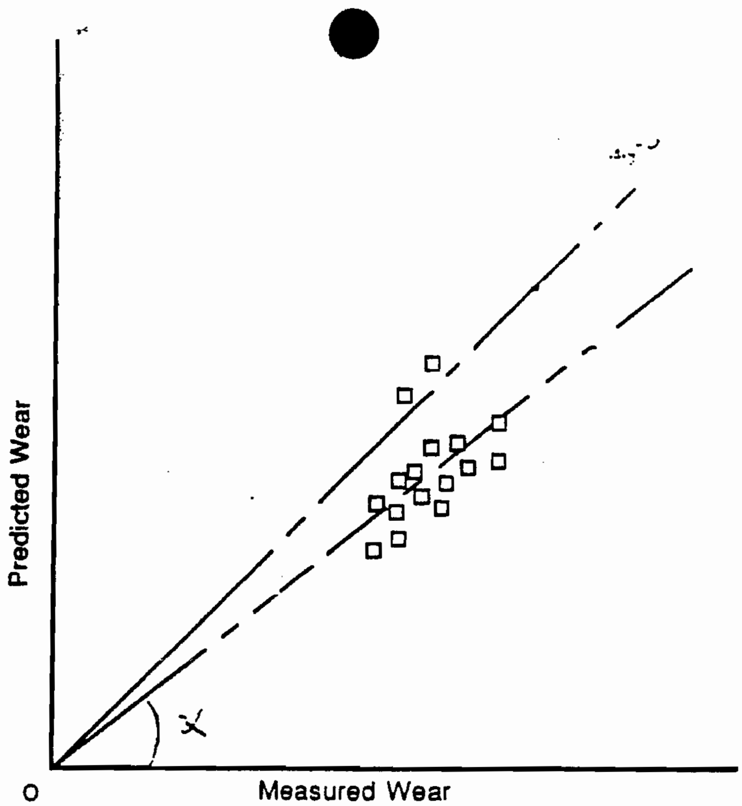
Measured and predicted data are plotted on a graph (Fig 1 and 2);

The best fitting line passing through the origin is drawn;

LCF is a cotangent of the angle (α) made by this line and the abscissa.

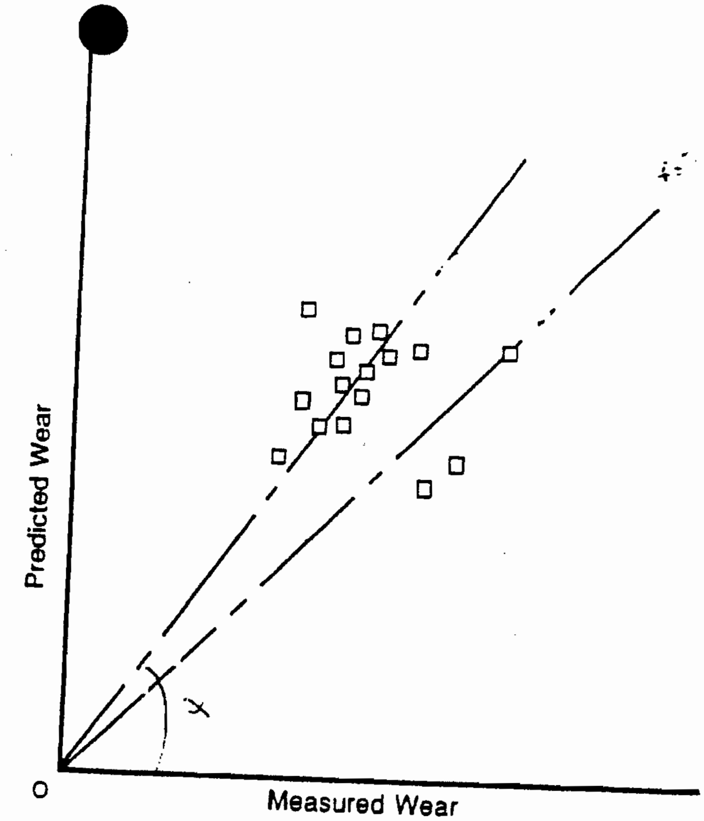
As long as the plant model remains unchanged, the LCF can be used for all subsequent calculations, even if the velocity or temperature change.

Step 3 Predicted Wall Thickness = LCF x Initial Wall Thickness (Fig 3)



Example of Under-Prediction

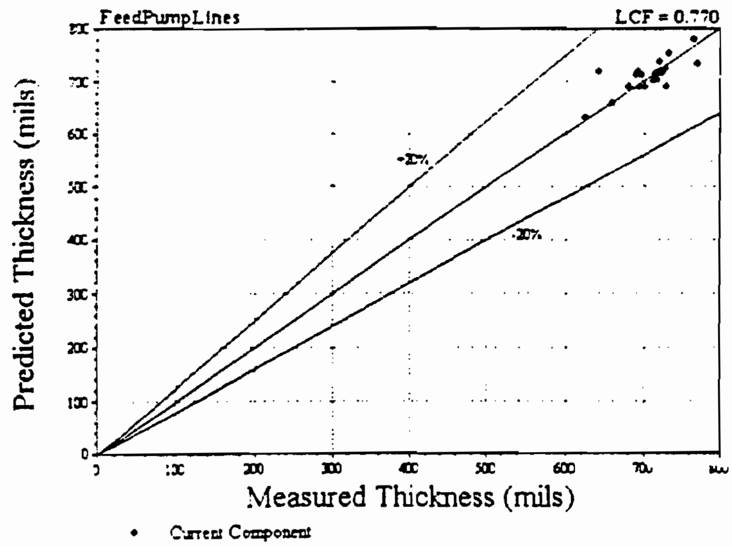
FIG. 1



Example of Over-Prediction

FIG. 2

LCF = Cotangent α



Predicted Thickness

FIG. 3

Request: How is the CHECWORKS code used to determine inservice inspection (ISI) periods?

Response:

CHECWORKS is not used for predicting ISI periods. However, it is used for predicting the components which will not meet wall thickness criteria before the next outage. For that purpose, during an outage, a prediction is made using CHECWORKS of wall thickness at the next outage for the components included in the code. The components which indicate that their wall thickness become less than required by specifications are repaired or replaced.

Request: Provide quantitative evidence from the staff to support the use of the Vessel Internals Project's evaluation criteria for SCC, recognizing that those criteria are based on data obtained at (very) low flow rate conditions in comparison to those proposed under uprate conditions.

Response:

[To be provided at a later date]

EMEB Responses to ACRS Subcommittee Requests on DAEC EPU

Request: Provide data to show that flow induced vibration is not a problem during EPU at DAEC.

Response:

Data from the licensee on jet pump vibration (see attached table) indicates that calculated results from extrapolated lower power test data are in good agreement with actual test data. For jet pump vibration at 120 percent power, the maximum resultant stress based on absolute sum of stresses in all modes between 23 to >200 Hz, is 87.5 percent of the allowable limit. The licensee indicated that all calculated flow induced vibration stresses are below the allowable limit. In addition, the potential for the flow induced vibration does not increase for components in the reactor vessel lower plenum and core region since there is no increase in the core flow. The Nuclear Regualtory Commission (NRC) staff concludes that the licensee's evaluation reasonably demonstrates that flow induced vibration is not a problem during EPU at DAEC.

Request: Discuss the basis for acceptability of the steam dryer against failure considering the increase in steam flow and the subsequent flow induced vibration as a result of EPU.

Response:

In its letter dated August 1, 2001, the licensee responded to the NRC staff's request for additional information with regard to the structural integrity of the steam dryer subject to the flow induced vibration following the proposed power uprate. With regard to the dryer, the NRC staff determined that the increase in steam flow will not have adverse effects on the steam dryers as a result of the proposed EPU. The steam dryer failures are not expected to occur for the following reasons:

(1) There is no significant increase in pressure, temperature, and flow that affect the dryer structural integrity.

(2) There have been no identified cracks or failures in the dryer during the plant operating history.

(3) Per BWRVIP-06, dryers are visually inspected during removal in each refueling outage, and any significant cracking will be identified and assessed.

(4) Stresses in the dryers during plant operation are small in relation to the allowable endurance limit of 10 ksi and significant cracking of the dryer during one operating cycle is considered very unlikely.

(5) The steam dryer is designed to maintain its structural integrity during the steam line break (e.g., stresses are below American Society of Mechanical Engineers faulted condition limits).

ATTACHMENT 2

Request: Discuss the acceptability of calculated cumulative usage factors (CUFs). Also, elaborate on decreases of certain CUF values in light of the expected increase in stresses from EPU.

Response:

The following factors contribute to the conservatism in the fatigue analysis:

(1) Use of the largest scale factor derived from changes of pressure, temperature, and flow for the EPU.

(2) Application of the scale factor to resultant stresses that include stresses not only due to pressure and temperature, but also due to seismic and mechanical loads not affected by the EPU.

(3) In general, use of minimum allowable cycles (N1) corresponding to the worst transient in calculating the CUF ((n1+n2+n3+...)/N1).

Since the methodology using the scale factor is conservative, calculated EPU CUFs that are less than the allowable limit of one, for the reactor vessel components (e.g., feedwater nozzle) are acceptable.

When a conservative estimate of EPU CUF is greater than 1.0 (e.g., at recirculation outlet nozzle), the licensee refined the calculation by using the sum of individual EPU CUF_i for each transient instead of using the minimum allowable cycles. Resultant CUF was considerably lower. This clearly demonstrates the inherent conservatism in the original calculation.

The NRC staff considers this approach to be reasonable and acceptable.

From: "Deann Raleigh" <draleigh@scientech.com>
To: "Michele Kelton" <MSK@nrc.gov>
Date: 6/9/04 12:34PM
Subject: RE: WHAT'S YOUR MAILING ADDRESS?

Thank you Michele,

If you could send to our Florida office, that would be great:

Theresa Sutter
SCIENTECH
2650 McCormick Drive STE 300
Clearwater, Florida 33759-1049

Thanks so much! I really appreciate it,

Deann

-----Original Message-----
From: Michele Kelton [mailto:MSK@nrc.gov]
Sent: Wednesday, June 09, 2004 11:20 AM
To: draleigh@scientech.com
Subject: WHAT'S YOUR MAILING ADDRESS?

I responded to Mr. Browning and will be sending you a "cc."

Michele
ACRS/ACNW

From: Ralph Caruso
To: Browning, Tony ; Michele Kelton
Date: 6/2/04 11:13AM
Subject: Re: Request for Documents

Tony,

I don't know if you remember me, but I am now working on the ACRS staff, and this request has come to me for resolution. Can you give me a phone call so we can talk this through? 301-415-8065.

Thanks.

Ralph Caruso

>>> "Browning, Tony " <Tony.Browning@nmcco.com> 05/28/04 01:24PM >>>
Ms. Kelton,

As way of background, I work for the Nuclear Management Co. (NMC) in the Regulatory Affairs Department at the Duane Arnold Energy Center (DAEC). I am currently compiling some background information on the Extended Power Uprate project for the DAEC and had requested support from Scientech - LIS in obtaining some historical documentation relating to the ACRS review of the Duane Arnold Extended Power Uprate. Specifically, the memoranda and attachments cited as References 8 and 10 in the ACRS letter ACRSR-1965, dated October 17, 2001.

Scientech - LIS suggested that I contact you directly about obtaining these documents. My understanding from LIS is that the attachments to this memo are labeled "proprietary." My expectation is that this information is proprietary to the General Electric Co. and was docketed by NMC in support of the DAEC Extended Power Uprate. Consequently, it is information to which we are already privy under our agreements with General Electric as their client. Thus, would you be able to release those documents directly back to us, along with the associated memoranda; or, will it be necessary for us to formally request them under FOIA? Please advise.

Thanks in advance,

Tony Browning
Principle Engineer - Regulatory Affairs
NMC - DAEC

From: "Browning, Tony " <Tony.Browning@nmcco.com>
To: <MSK@nrc.gov>
Date: 5/28/04 1:24PM
Subject: Request for Documents

Ms. Kelton,

As way of background, I work for the Nuclear Management Co. (NMC) in the Regulatory Affairs Department at the Duane Arnold Energy Center (DAEC). I am currently compiling some background information on the Extended Power Uprate project for the DAEC and had requested support from Scientech - LIS in obtaining some historical documentation relating to the ACRS review of the Duane Arnold Extended Power Uprate. Specifically, the memoranda and attachments cited as References 8 and 10 in the ACRS letter ACRSR-1965, dated October 17, 2001.

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Thanks in advance,

Tony Browning
Principle Engineer - Regulatory Affairs
NMC - DAEC

From: Michael Snodderly
To: Michele Kelton
Date: 5/26/04 5:23PM
Subject: Fwd: ACRS Letter re Duane Arnold PUR

Please respond to Ms. Raleigh per our conversation. You may want to file the October 3, 2001 memo chronologically.

Thanks again,

Mike

CC: Ralph Caruso

Mail Envelope Properties (40B50AB9.601 : 19 : 52537)

Subject: Fwd: ACRS Letter re Duane Arnold PUR
Creation Date: 5/26/04 5:23PM
From: Michael Snodderly

Created By: MRS1@nrc.gov

Recipients
twf1_po.TWFN_DO
MSK (Michele Kelton)
RXC CC (Ralph Caruso)

Post Office	**Route**
twf1_po.TWFN_DO	

Files	**Size**	**Date & Time**
Mail		
MESSAGE	705	05/26/04 05:23PM

Options
Expiration Date: None
Priority: Standard
Reply Requested: No
Return Notification: None

Concealed Subject: No
Security: Standard

From: "Deann Raleigh" <draleigh@scientech.com>
To: <pab2@nrc.gov>, <mrs1@nrc.gov>
Date: 5/26/04 4:09PM
Subject: ACRS Letter re Duane Arnold PUR

Dear Mike and Paul,

I've been trying to come up with some ACRS documents that I cannot find in ADAMS nor the NRC website. Can either of you help?

I am looking for the memo (P. Boehnert to ACRS), and its attachments, dated October 3, 2001, referred to in the attached ACRS letter as References 8 & 10.

Thanks!

Best Regards,

Deann Raleigh
Licensing Information Service
SCIENTECH, Inc.
240-626-9556

ACRSR-1965

October 17, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: DUANE ARNOLD ENERGY CENTER EXTENDED POWER UPRATE

Dear Chairman Meserve:

During the 486th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2001, we met with representatives of the NRC staff and the Nuclear Management Company to review the license amendment request for an increase in core thermal power for the Duane Arnold Energy Center (DAEC), pursuant to the General Electric Nuclear Energy Extended Power Uprate Program. Our subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during meetings held on June 12 and September 26-27, 2001. During our review, we had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The DAEC application for the extended power uprate should be approved.

2. The Safety Evaluation Report (SER) should be revised to document adequately the technical resolution of the issues raised by the staff.

3. The staff should develop improved guidance on the detail to be provided in SERs and criteria for when independent assessments should be performed to complement its reviews of applicant submittals.

DISCUSSION

The Nuclear Management Company has requested an amendment to the DAEC operating license for a 15.3% increase over the plant's current operating power limit. Previously, the staff had approved a smaller power uprate. Consequently, the current application is for a power uprate of 20% over the originally licensed power. This is the largest power uprate ever considered for boiling water reactors (BWRs) in the United States. It is anticipated that many other licensees will request similarly large increases in the operating powers of BWRs. Consequently, we anticipate that staff review of the DAEC power uprate will be a template for future reviews and will set the expectations for many future power uprate applications.

A generic methodology for evaluating and justifying power uprates of up to 20% for BWRs has been developed by General Electric. This generic methodology has been approved by the staff. The DAEC application has adopted this methodology and, in fact, the NRC staff has used the methodology to guide its review of this power uprate application.

The power increase at DAEC will be achieved by increasing steam production, while holding liquid flow in the core, dome pressure and temperatures quite near current values. The increased steam production is achieved by "flattening" the core power profile, which involves increasing power generation in the outer regions of the core. There is an increase in feedwater flow to match the increased production of steam. Balance-of-plant modifications are required and will cause the DAEC power increase to be performed in two steps.

Many technical issues must be addressed in an application for power uprate. Of these, we consider five to be especially significant:

1. Susceptibility of the plant to ATWS (Anticipated Transients Without Scram)

2. ATWS recovery

3. Reduction in some of the times available for operator actions because of higher decay heat

4. Material degradation due to irradiation-assisted stress corrosion cracking (IASCC) of reactor internals and flow-assisted corrosion and fatigue of feedwater piping

5. Containment response to accident events involving higher decay heat levels

Our examinations of the staff's SER and Requests for Additional Information submitted by the staff to the applicant persuaded us that the staff had raised numerous, pertinent issues concerning the conformance of the power uprate to approved methodologies. Though we persuaded ourselves eventually that the DAEC power uprate could be accomplished safely, we found it difficult to obtain information on the technical resolution of the issues either in the staff's SER or in our meetings with the staff. An exception to this common difficulty was the resolution of issues concerning containment response to design-basis accident events. In this case, the staff provided us a report on comparisons of applicant analyses with analyses done using an independent computational tool.

We found it far more difficult to assure ourselves that the DAEC core is susceptible only to global power oscillations and does not need to consider local power oscillations. It was similarly difficult to assure that ATWS recovery methods were applicable to cores with flattened power profiles, that critical human actions had been identified with adequate independence by the staff, and that material degradation sensitivities had been adequately assessed.

Many of the challenges that we encountered in our review of the DAEC power uprate application could have been eased if the staff had improved guidance on the detail to be provided in SERs and developed criteria for when independent assessments should complement reviews of applicant submittals.

ACRS Members Mario Bonaca and F. Peter Ford did not participate in the Committee's review of this matter.

Sincerely,

/RA/

George E. Apostolakis
Chairman

References:

1. Memorandum dated September 5, 2001, to John T. Larkins, ACRS, from J. Zwolinski, Office of Nuclear Reactor Regulation, NRC, Subject: Draft Safety Evaluation for Duane Arnold Energy Center Extended Power Uprate (draft Predecisional report).
2. GE Nuclear Energy, Topical Report, NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," February 1999 (Proprietary).
3. GE Nuclear Energy, Topical Report, NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," February 2000 (Proprietary)
4. GE Nuclear Energy, Topical Report, NEDC-32523P-A, Supp 1, Volume 1, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate - Supplement 1, Volume I," February 1999, and Volume II, April 1999 (Proprietary).
5. GE Nuclear Energy, Topical Report, NEDC-32992P, "ODYSY Application for Stability Licensing Calculations," October 2000 (Proprietary).
6. BWR Owners' Group Letter dated March 8, 1996, transmitting GE Nuclear Energy Licensing Topical Report, BWR Owners Group Long-Term Stability Solutions Licensing Methodology, NEDO-31960-A, November 1995.
7. Report (draft final) from A. Cronenberg, ACRS, "Margin Reduction Estimates for Re-Licensed/Uprated Plants: Hatch Case Study," August 2001.
8. Response by Nuclear Management Company to ACRS Thermal-Hydraulic Phenomena Subcommittee question, undated, attached to October 3, 2001 Memorandum from P. Boehnert to ACRS Members.
9. U.S. Nuclear Regulatory Commission, Technical Evaluation Report, ISL-NSAD-NRC-01-001, "Duane Arnold Energy Center Extended Power Uprate Containment Analysis Audit Calculation," B. Gitnick, Information Systems Laboratory, Inc., July 2001.
10. Memorandum (undated) from J. Zwolinski, Office of Nuclear Reactor Regulation, NRC, Subject: Responses to Advisory Committee on Reactor Safeguards (ACRS) Subcommittee Questions Regarding Duane Arnold Energy Center Extended Power Uprate, attached to October 3, 2001, Memorandum from P. Boehnert, to ACRS Members (contains Proprietary information).
11. GE Nuclear Energy Licensing Topical Report, NEDC-32980P, Rev. 1, "Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate," April 2001 (Proprietary).
12. Nuclear Management Company Memorandums: Response to Request for Additional Information - Duane Arnold Energy Center Extended Power Uprate, dated April 9, March 23, April 16, April 16 (Proprietary), May 8 (Proprietary), May 10, May 11, May 11 (Proprietary), May 22, May 29 (Proprietary),and June 5, 2001.
13. Nuclear Management Company Memorandums: Response to Request for Additional Information - Extended Power Uprate, June 11, June 18, June 21, June 28, July 11, July 19, July 25, August 1 (proprietary), August 1(proprietary), August 10 (proprietary), August 16 (proprietary), and August 21, 2001.