EVALUATION OF CHANGE PURSUANT TO 10 CFR 50.92

Background:

In 1991, an independent evaluation of the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC) was conducted as part of the DAEC TS Improvement Program. A portion of the Program included comparison of the DAEC TS with TS from similar plants, Standard TS (NUREG-1202, July 1986), and the draft Improved Technical Specifications (NUREG-1433). Based on this comparison, the current DAEC TS Section 3.6, "Primary System Boundary," has been rewritten and is the subject of this submittal. Specifically, the proposed changes contained in this submittal will revise the Limiting Conditions For Operation (LCO) and Surveillance Requirements (SR) for Thermal and Pressurization Limitations, Coolant Chemistry, Coolant Leakage, Safety and Relief Valves, Jet Pumps, Jet Pump Flow Mismatch, Structural Integrity, and Shock Suppressors (Snubbers). In addition, definitions for IDENTIFIED, UNIDENTIFIED LEAKAGE, TOTAL LEAKAGE, and DOSE EQUIVALENT I-131 are being incorporated into TS Section 1.0, "Definitions."

Iowa Electric Light and Power Company, Docket No. 50-331

Duane Arnold Energy Center, Linn County, Iowa

Date of Amendment Request: December 31, 1992

Description of Amendment Request:

The proposed amendment revises DAEC TS Sections 1.0 and 3.6 to provide additional definitions and improve the clarity and consistency of LCOs and SRs for Primary System Boundary. The majority of the changes being proposed are consistent with comparable Specifications in the Standard TS (NUREG-1202). The other changes are editorial or administrative in nature.

Definition 41, IDENTIFIED LEAKAGE

The existing DAEC TS do not define IDENTIFIED LEAKAGE. This definition is being added to improve clarity and consistency with LCO 3.6.C, "Coolant Leakage." This definition is consistent with the guidance provided by Standard TS.

Definition 42, TOTAL LEAKAGE

The existing DAEC TS do not define TOTAL LEAKAGE. This definition is being added to improve clarity and consistency with LCO 3.6.C, "Coolant Leakage."

Definition 43, UNIDENTIFIED LEAKAGE
The existing DAEC TS do not define UNIDENTIFIED LEAKAGE. This definition is being added to improve clarity and consistency with LCO 3.6.C, "Coolant Leakage." This definition is consistent with the guidance provided by Standard TS.

Definition 44, DOSE EQUIVALENT I-131
The existing DAEC TS do not define DOSE EQUIVALENT I-131. This definition is being added to improve the clarity and consistency with LCO 3.6.B, "Coolant Chemistry." This definition is consistent with the guidance provided by the Standard TS.

TS Section 3/4.6, Primary System Boundary
The existing Applicability and Objective sections are editorially revised to use initial caps for Reactor Coolant System. This is consistent throughout the Section 3.6 rewrite.

TS Section 3/4.6.A, Thermal and Pressurization Limitations
This section is being revised to improve clarity and provide consistency within the DAEC TS as well as to adopt the specific language of the Standard TS. The existing TS do not specify clearly and concisely the actions to be taken when a specific LCO or SR is exceeded. The proposed revisions to this section eliminate that problem. A summary of the proposed changes follows:

* Existing TS Section 3.6.A.1 has not been changed.

* Existing TS Section 3.6.A.2 is being revised to delete the last part of the LCO which provides a commitment to update Figure 3.6-1 six months prior to 16 effective full power years. This commitment is not required in the LCO since it does not verify system OPERABILITY requirements or provide any additional information assisting in the operation of the plant or mitigating any accidents. This commitment has been relocated to the Bases Section.

* Existing TS Section 3.6.A.3 is being editorially revised to identify the location where temperature readings are to be taken before reactor vessel head bolting studs are placed under tension. In addition, editorial changes are being made to be consistent with the existing Bases.

* Proposed TS Section 3.6.A.4 is being added. The existing TS do not specify the actions if the temperature/pressure limits are exceeded. The addition of this LCO provides time limits for bring temperature/pressure

back within specification requirements and performing an engineering evaluation. It requires shutdown only if the plant cannot comply with the specific actions.

* Existing TS Section 3.6.A.5 has been revised to provide additional guidance, consistency, and to incorporate specific information from the Standard TS. The LCO will specifically identify the MODES of operation which apply to the recirculation pump. The current TS did not provide this information. Existing TS Sections 3.6.A.4 and 3.6.A.5 were editorially revised to provide clarity and consistency within the DAEC TS using the guidance provided using the Standard TS and combined into a single section.

* Existing SR 4.6.A.1 has been editorially revised to provide consistency and clarity within this section of the DAEC TS. In addition, the word "logged" has been replaced with the word "recorded." This is discussed in more detail in Attachment 2.

* Existing SR 4.6.A.2 has been revised to delete the last two paragraphs which discuss when the last specimens were withdrawn and when the next ones are scheduled to be withdrawn. This type of information should not be contained in the SR or LCO. This information has been incorporated into the Bases Section. A SR has been added which requires specimens to be removed in accordance with 10CFR50, Appendix H. This SR is in accordance with Generic Letter 91-01, "REMOVAL OF THE SCHEDULE FOR THE WITHDRAWAL OF REACTOR VESSEL MATERIAL SPECIMENS FROM TECHNICAL SPECIFICATIONS."

* Existing SR 4.6.A.3 has not been changed.

* Existing SRs 4.6.A.4 and 4.6.A.5 have been reorganized. The existing information has been maintained and itemized under the proposed SR 4.6.A.4.

TS Section 3/4.6.B, Coolant Chemistry

This section is being revised to provide clarification and consistency within the DAEC TS as well as adopt specific language of the Standard TS. The entire existing Section 3/4.6.B is being either revised or new LCOs and SRs added. A summary of changes are as follows:

* Existing TS LCO 3.6.B.1 has been revised and divided into three different LCOs. The existing LCO contained information which was difficult to read and understand. This proposed revision does not change the actual intent of the existing LCO but made it clearer.

* Existing TS LCO 3.6.B.2.a is being revised. The existing TS contains information that will be easier to understand in tabular format. In addition, the LCO references a steaming rate of 100,000 pounds per hour. This LCO was revised to place appropriate information in a new table which references a temperature associated with a MODE of operation. This is more meaningful to plant personnel than rates in "pounds per hour." The other information contained in this LCO has been relocated in other TS within this section.

* Existing TS LCO 3.6.B.2.b has been revised. Much of the information contained is incorporated in new Tables 3.6.B.2-1 and 4.6.B.1-1. As stated above, more meaningful plant MODE conditions have been used instead of rates in "pounds per hour."

* Existing TS LCO 3.6.B.2.c has been incorporated into proposed LCOs within this section.

* Existing TS LCO 3.6.B.2.d has been incorporated into several proposed LCOs within this section.

* Existing TS LCO 3.6.B.3.a has been revised. This information is either provided in the proposed LCOs or in the new Table 3.6.B.2-1.

* Existing TS LCO 3.6.B.3.b has been revised and incorporated into proposed SR 4.6.B.2.e and 4.6.B.2.f. This requires the conductivity recording on a continuous basis. If the monitor is inoperable, an in-line sample is taken and evaluated. The monitors are to be channel checked with an in-line flow cell at least once every 7 days.

* Existing TS LCO 3.6.B.4 has been incorporated into the individual proposed LCOs.

* Proposed TS LCO 3.6.B.2.d has been added, requiring an engineering evaluation be performed to verify structural integrity if the limits specified in Table 3.6.B.2-1 are exceeded.

* Existing SRs 4.6.B.1.a through 4.6.B.1.g have been revised and incorporated into proposed Table 4.6.B.1-1 and the proposed SRs. Editorial changes have also been made with the SRs patterned after those of the Standard TS.

* Existing SR 4.6.B.2, 4.6.B.2.a, and 4.6.B.2.b have been incorporated into proposed Table 4.6.B-1 and the proposed SRs. Editorial changes have been made to the SRs to be consistent with Standard TS.

* Proposed TS SRs 4.6.B.2.a, 4.6.B.2.b, 4.6.B.2.c, and 4.6.B.2.d have been added to provide additional guidance for obtaining samples as specified in Table 3.6.B.2-1.

* Existing SRs 4.6.B.3.a and 4.6.B.3.b have been revised and retained as proposed SR 4.6.B.2.e and 4.6.B.2.f. The specific monitoring locations have been relocated to the Bases Section.

TS Section 3/4.6.C, Coolant Leakage

This section is being revised to clarify existing LCOs, SRs, add specific shutdown requirements and provide consistency with the DAEC TS and by Standard TS. A summary of the proposed changes are as follows:

* Existing TS LCOs 3.6.C.1, 3.6.C.1.a, 3.6.C.1.b, and 3.6.C.1.c have been editorially revised to provide clarity. The editorial changes consist of capitalizing defined terms and replacing existing words to be consistent with Standard TS.

* Existing TS LCO 3.6.C.3 has been renumbered to LCO 3.6.C.2. Proposed TS LCO 3.6.C.2 has been editorially revised to provide clarity and consistency with the DAEC TS and the guidance provided in the Standard TS. In addition, a 4 hour action statement has been added. This will allow leakage to be brought back within its limits before a shutdown action is initiated. The proposed shutdown action has also been revised to incorporate the guidance provided in the Standard TS.

* Existing LCO 3.6.C.2 has been renumbered to 3.6.C.3. Existing LCO 3.6.C.2 does not provide either a reference or specific requirements that define Sump System OPERABILITY. Therefore, proposed TS LCO 3.6.C.3 is being revised to reference the applicable section of the DAEC TS Table 3.2-E which defines Sump System OPERABILITY.

* Proposed TS LCO 3.6.C.4 is being added to state specific actions to take in the event that the Sump System is inoperable. In addition, a shutdown requirement is being added to require being in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours if the Sump System cannot be restored to OPERABLE status within 24 hours. This addition is consistent with the guidance provided in the Standard TS.

* Proposed TS LCO 3.6.C.5 is being added to state specific actions and shutdown requirements to take in the event neither the Sump System nor the Air Monitoring System is OPERABLE. This revision is consistent with the guidance provided in the Standard TS.

* Existing SR 4.6.C.1 is being revised to use initial capital letters for the Reactor Coolant System and Sump System. This is an editorial change which is consistent with the rest of the DAEC TS.

* Proposed TS SR 4.6.C.2 is being added to verify OPERABILITY of the Sump System in accordance with Table 4.2-E. The existing TS SRs do not currently contain this requirement.

* Existing SR 4.6.C.2 is being renumbered to 4.6.C.3. The existing SR does not define requirements to verify Air Sampling System OPERABILITY in the event that the Sump System becomes inoperable. The revision to existing SR 4.6.C.2 (now proposed SR 4.6.C.3) consists of verifying the Air Sampling System is OPERABLE in accordance with Table 4.2-E.

* The Bases Section 3.6.C & 4.6.C have been revised to reflect the proposed changes.

<u>TS Section 3/4.6.D, Safety and Relief Valves</u>
This section has been revised to clarify existing LCOs, SRs, add specific shutdown requirements, and to provide consistency with the DAEC TS and the guidance provided by the Standard TS. A summary of the proposed changes are as follows:

* Existing TS LCO 3.6.D.1 is being revised to use proper MODE titles. For consistency, all defined terms are to be identified in the DAEC TS in all caps. In addition, a note was added to state that SRVs which perform an ADS function must also satisfy the OPERABILITY requirement as specified in Specification 3.5.F.

* Existing TS LCO 3.6.D.2.a is being revised to clarify the LCO requirements in the event that the safety function of one relief valve becomes inoperable.

* Existing TS LCO 3.6.D.2.b is being revised to clarify the LCO requirements in the event that the safety function of two relief valves become inoperable.

* Existing TS LCO 3.6.D.3 is being revised to clarify and state the shutdown requirements when TS LCO 3.6.D.1 or 3.6.D.2 is not complied with.

* Existing TS SR 4.6.D.1 is being revised to be more consistent with the guidance provided in the Standard TS. The revision also clarifies the specific requirements for pressure testing, removal, and replacement for safety and relief valves.

* Existing TS SR 4.6.D.2 is being editorially revised by capitalizing "OPERATING CYCLE" since it is a defined term in the DAEC TS.

* Existing TS SR 4.6.D.3 is being editorially revised by capitalizing "OPERATING CYCLE" since it is a defined term in the DAEC TS. In addition, the footnote is being deleted as it is a superfluous statement.

* Existing TS SR 4.6.D.4 is being revised editorially by replacing the word "required" with the word "specified". This change is consistent with the DAEC TS and the guidance provided by Standard TS.

* The Bases Section 3.6.D & 4.6.D have been revised to reflect the proposed changes.

<u>TS Section 3/4.6.E, Jet Pumps</u>
This section has been revised to clarify existing LCOs, SRs, add specific shutdown requirements, and to provide consistency within the DAEC TS as well as with the guidance provided by the Standard TS. A summary of the proposed changes are as follows:

* Existing TS LCO 3.6.E.1 is being revised to refer to defined MODES of operation. LCO 3.6.E.1 also contains a statement that, if a specific surveillance cannot be met, an additional surveillance is to be performed within 24 hours. This proposed Amendment relocates this information in its entirety to proposed SR 4.6.E.1.c.

* Existing TS LCO 3.6.E.1.a and 3.6.E.1.b have been revised and renumbered to proposed LCO 3.6.E.1.a, 3.6.E.1.a.1, and 3.6.E.1.a.2. These proposed changes are being made to provide clarity within the LCO.

* Proposed TS LCO 3.6.E.1.a.2 has been editorially revised. In addition, a shutdown requirement has been proposed to be consistent with the guidance provided by the Standard TS and to eliminate unnecessarily cycling the plant to the COLD SHUTDOWN condition as currently required in the DAEC TS.

* Editorial changes are made in existing TS SR 4.6.E.1. The word OPERABILITY is a defined term in the DAEC TS and is to appear in capital letters. Instead of abbreviating recirculation, the proposed change spells it out.

* Existing TS SR 4.6.E.1.a and 4.6.E.1.b have minor editorial changes made as noted in Attachment 2 providing consistency throughout the TS.

* Proposed TS SR 4.6.E.1.c has been moved from existing LCO 3.6.E.1 as discussed above.

* Existing TS SR 4.6.E.2 has not been changed.

* Existing TS SR 4.6.E.3 has been editorially changed for clarity.

* Existing TS SR 4.6.E.4 has not been changed.

TS Section 3/4.6.F, Jet Pump Flow Mismatch

This section has been revised to clarify existing LCOs, SRs, add specific shutdown requirements and to provide consistency within the DAEC TS and the guidance provided by the Standard TS. A summary of the proposed changes are as follows:

* Existing TS LCO 3.6.F.1 has been divided into two itemized sections proposed as TS LCOs 3.6.F.1 and 3.6.F.2. Minor editorial changes were made to each LCO in order to allow it to stand alone. These minor changes do not change the intent or requirements of the existing LCO.

* Proposed TS LCOs 3.6.F.3 and 3.6.F.3.a have been added as clarification and for consistency with the guidance provided by the Standard TS. The addition of this LCO allows 2 hours for the recirculation pump speeds to be restored within the above limits. The current TS does not allow any time to restore the system to within the limits before taking further action.

* Existing TS LCO 3.6.F.2 was revised and renumbered as proposed LCO 3.6.F.3.b.

* Existing SR 4.6.F.1 has been editorially revised to provide additional clarification and consistency by replacing the words "checked and logged" with "verified."

* Existing SR 4.6.F.2 has been editorially revised changing the word "Specification" to "Surveillance Requirement." The number referenced is a Surveillance Requirement number and is identified accordingly.

TS Section 3/4.6.G, Structural Integrity

This section has been revised to clarify existing LCOs, SRs, and specific shutdown requirements, add LCOs, and provide consistency with the DAEC TS and the guidance provided by Standard TS. A summary of the proposed changes are as follows:

* Existing TS LCO 3.6.G.1 has been revised to provide clarity by specifically identifying when structural integrity is required and to also correct the reference to ASME Section XI Code Class 1, 2, and 3 components.

* Proposed TS LCO 3.6.G.2 has been added, providing specific actions for Class 1 and Class 2 components when they do not conform to the ASME Section XI requirements. This proposed change is a clarification in that the existing TS does not provide specific actions if the Class 1 or Class 2 component does not meet TS LCO 3.6.G.1. The proposed wording is consistent with the guidance provided in the Standard TS.

* Proposed TS LCO 3.6.G.3 has been added, providing specific actions for Class 3 components when they do not conform to the ASME Section XI requirements. This proposed change is a clarification in that the existing TS does not provide specific actions if the Class 3 component does not meet TS LCO 3.6.G.1. The proposed wording is consistent with the guidance provided in the Standard TS.

* Proposed TS LCO 3.6.G.4 has been added. The existing TS do not include actions to be taken in the event that a Class 1, 2, or 3 component(s) cannot meet the structural integrity requirements when above 212°F.

* Existing TS SR 4.6.G.1 has been revised to include testing requirements of Class 1, 2, and 3 pumps and valves. These requirements were incorporated into this SR from existing SR 4.6.G.2, which will be deleted.

* Existing TS SR 4.6.G.1.a is being deleted. The information contained in the existing SR does not provide any guidance or verification of equipment OPERABILITY. This information is already included in the Bases Section.

* Existing TS SR 4.6.G.2 is being deleted. The requirements for pump and valve testing are being relocated to TS LCO 3.6.G.1 above.

* Existing TS SR 4.6.G.2.a is being deleted. The information contained in the existing SR does not provide any guidance or verification of equipment OPERABILITY. This information is already included in the Bases Section.

* Existing TS SR 4.6.G.3 has been revised and renumbered to proposed SR 4.6.G.2. The word "augmented" replaced the word "inservice" which is a more grammatically correct and accurate description of DAEC's program.

TS Section 3/4.6.H, Shock Suppressors (Snubbers)

This section has been revised to clarify existing LCOs, SRs, and provide consistency within DAEC TS and the guidance provided by the Standard TS. A summary of the proposed changes follows:

* Existing TS LCO 3.6.H.1 is being revised to state the specific MODES and conditions in which the LCO is applicable, capitalize OPERABLE, and provide other editorial changes.

* Existing TS LCO 3.6.H.2 is being editorially revised to correct the Specification number referenced and to abbreviate Limiting Conditions For Operation (LCO) as it normally appears.

* Add Table 4.6.H-1 to the existing TS. This Table is being added to provide requirements for snubber visual inspection intervals for the number of unacceptable snubbers. This revision is being made as a result of NRC Generic Letter 90-09, "Alternate Requirements For Snubber Visual Inspection Intervals and Corrective Actions."

* Existing SR 4.6.H is being editorially revised by inserting the word "augmented" to clarify that the DAEC is an augmented inspection program. In addition, references to Surveillance Requirements 4.6.H.5 and 4.6.H.6 are being added.

* Existing SR 4.6.H.1 for visual inspections is being revised. The revision is being made to ensure that the DAEC TS comply with NRC Generic Letter 90-09.

* Existing SR 4.6.H.2 is being revised to conform to the guidance provided by Standard TS. The language of the Standard TS is clearer and provides expanded and specific requirements for determining the next inspection interval for unacceptable snubbers. In addition, the proposed SR requires a review and evaluation be performed and documented to justify continued operation with an unacceptable snubber.

* Proposed SR 4.6.H.3, "Transient Event Inspection" is being added. The existing SRs do not have this section. The addition of this section provides specific guidance in the event that a potentially damaging transient occurs. If one does occur, the new SR requires that a review of operational data or a visual inspection of the system(s) be performed. The addition of this SR is consistent with the guidance provided by the Standard TS.

* Existing SR 4.6.H.3 is being changed to proposed SR 4.6.H.4. In addition, "OPERATING CYCLE" is being changed to all caps. OPERATING CYCLE is a

defined term and is to appear in all caps. The word "Specification" is being changed to "Surveillance Requirements." The referenced numbers in this SR are being renumbered due to the addition of new SRs. The footnote is also being deleted. 1t contains superfluous information that is not needed to perform any SR or LCO action.

* Existing SR 4.6.H.4 is being changed to proposed SR 4.6.H.5. In addition there were some editorial changes made. This makes the proposed SR 4.6.H.5 consistent with proposed SR 4.6.H.6.

* Existing SR 4.6.H.5 is being changed to proposed SR 4.6.H.6. This is an editorial change.

* Add SR 4.6.H.7, "Functional Testing of Repaired and Replaced Snubbers." The Addition of this SR ensures that the repair or replacement of snubbers shall meet the functional test criteria before installation in the unit.

* Add SR 4.6.H.8, "Snubber Service Life Replacement Program." This SR ensures that the service life of the snubbers is monitored, ensuring that the service life is not exceeded between surveillance inspections.

* Existing SR 4.6.H.6 is being deleted. The intent of the requirements for this surveillance are incorporated into proposed SR 4.6.H.7 and SR 4.6.H.8.

* The Bases Section 3.6.H & 4.6.H have been revised to reflect the proposed changes.

Basis for proposed no significant hazards consideration determination:
The Commission has provided standards (10CFR50.92 (c)) for determining whether a significant hazards consideration exists. A proposed amendment to the facility's operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

THERMAL AND PRESSURIZATION LIMITATIONS

All components of the Reactor Coolant System (RCS) are designed to withstand effects of cyclic loads due to system Temperature/Pressure (T/P) changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. The LCO limits the T/P changes during heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figure 5.3-1 of the DAEC UFSAR, shows three operating limit curves, including irradiation shift of the core beltline region curves to their position at end of life (32 full power years). The three curves represent three specific conditions: a) system hydrostatic and leakage tests, b) non-nuclear heatup or cooldown and low level physics tests, and c) core critical operation. The curves were established by requirements of Section III, Appendix G, of the ASME Code and by 10CFR50, Appendix G.

Each T/P limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when T/P indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the Reactor Coolant Pressure Boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

The T/P limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. Across the span of the T/P limit curves, different locations are more restrictive, and, thus the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner wall.

A possible consequence of violating the LCO limits is that the RCS is operated under conditions that could have resulted in brittle failure of the RCPB, possibly leading to a non-isolable leak or loss-of-coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

Comparison of the pressure versus temperature limits in UFSAR Figure 5.3-1 with intended normal and upset operating conditions, shows that the limits will not be exceeded during any foreseeable operating condition. Reactor operating procedures have been established such that actual transients will not be more severe than those for which the vessel was designed. Of the design transients, an upset condition produces the most adverse temperature and pressure condition with a minimum fluid temperature of 250°F and a maximum pressure peak of 1180 psig. Scram automatically occurs with initiation of this event, prior to the reduction in the fluid temperature, so the applicable operating limits are given by UFSAR Figure 5.3-1 curves A and A'. For a temperature of 250°F, the maximum allowable pressure at end of life exceeds 1400 psig for the intended margin against nonductile failure. The maximum transient pressure of 1180 psig is therefore within the specified allowable limits.

The average rate of reactor coolant temperature change during normal heatup and cooldown is limited by operating procedures to 100°F in any one hour period. During emergency and faulted conditions, the cooling rates may exceed this value as a result of rapid blowdown due to postulated valve malfunction or rupture accidents. The operator can compare the actual heatup and cooldown thermal and pressure cycle history for any given period of actual plant operating time with the reactor vessel cyclic design bases. This comparison will give, at any desired time, the status of actual vessel cyclic history and design cyclic requirements.

The revision discussed above is editorial in nature. The existing information does not provide the control room operator with any prudent action or guidance in the operation of the plant, mitigation of any accident, nor does it affect any procedural steps in the Emergency Operating Procedures. The proposed revision will not result in any loss of regulatory control since DAEC still meets the requirements specified in 10CFR50, Appendix H.

The proposed LCOs and SRs provide additional guidance, clarification, and consistency within the DAEC TS as well as utilizing the guidance provided by the Standard TS. The existing DAEC TS do not provide specific actions in the event the temperature/pressure limits are exceeded. The proposed LCO would allow 30 minutes to restore temperature/pressure limits. Once restored, an engineering evaluation is to be performed to determine any effects of the out-of-limit condition on the structural integrity of the RCS. If no effects are identified, operation is continued. If any of the above actions cannot be complied with, a reactor shutdown is initiated. Most violations of the temperature/pressure limits will not be severe, and the activity can be accomplished in a controlled manner. Besides restoring operation within limits, an evaluation is required to determine if the RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

Detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these transient are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started ensures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

Heating and cooling transients throughout plant life at uniform rates of 100°F/hr were considered in the temperature range or 100°F to 546°F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Code (1971 Edition including Summer 1972 Addenda).

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses that exceed ASME Code, Section III limits when the temperature differential is not greater than 145°F.

The minimum temperature of the fluid retained by a component can be used as a conservative estimate of metal temperature in evaluating the margin from the temperature at which the NDT properties were measured. Additional margin can usually be shown by calculating the temperature of the metal for the condition and area of concern.

The addition of the SR implementing Generic Letter 91-01,"REMOVAL OF THE SCHEDULE FOR THE WITHDRAWAL OF REACTOR VESSEL MATERIAL SPECIMENS FROM TECHNICAL SPECIFICATIONS" is within the requirements, approval, and guidance provided by the NRC. The addition of this SR does not involve a significant increase in the probability or consequences of an accident previously evaluated. This statement is based on the fact that the regulatory requirement of 10CFR50, Appendix H will remain in effect in the TS. Therefore, removal of any references to the specimen withdrawal will not result in any loss of regulatory control since any changes to this schedule are controlled by the requirements of 10CFR50 Appendix H.

Based on the addition of the previously mentioned SRs, several of the existing SRs are either being revised or deleted. These proposed changes are considered to be editorial in nature. These proposed changes are being made based on applying human factors concepts to minimize the potential for confusion, provide additional guidance not specifically provided in the previous TS, and provide consistency within the DAEC TS and the guidance as provided by the Standard TS. This transient is a nonlimiting event based on:

1. General Electric Standard Application for Reactor Fuel-United States Supplement, NEDO-24011-P-A-US, and

2. Duane Arnold Energy Center Single-Loop Operation, NEDO-24272.

The proposed changes greatly enhance the safety significance of the existing TS. These changes do not impact the safety analysis or any calculations or parameters utilized in the licensing bases for DAEC. In addition, the proposed changes do not relax any NRC regulations as contained in 10CFR50. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

COOLANT CHEMISTRY

The LCOs and SRs have been revised by placing in proposed Tables or by formatting in accordance with the guidance provided by the Standard TS. These revisions improve clarity, are consistent with the current industry practices, and provide additional guidance not specifically stated in the existing DAEC TS. The proposed changes do not change any safety analysis, parameters used in developing the safety analysis or any intended function for any safety related equipment.

COOLANT LEAKAGE

Reliable means are provided to detect leakage from the nuclear system barrier inside the drywell. Limits are established for nuclear system leakage rates so that appropriate action can be taken before the integrity of the nuclear system process barrier is unduly compromised.

The DAEC design includes a nuclear system leak detection, isolation, processing, and makeup system. This system provides for leakage control capability. The capability of this system is discussed in Section 3.6.C of this submittal.

The functions of the leak detection system are accomplished under normal operation or postaccident conditions so that normal (10CFR20) or accident (10CFR100) offsite dose limits do not exceed established values and in a manner in which the core and the containment cooling continuity is not impaired.

The leakage considered here is limited to that water or steam released from the nuclear system process barrier inside the primary containment. Leakage inside the drywell is treated separately from leakage elsewhere in the plant because the drywell contains a high concentration of nuclear system piping and is totally inaccessible during reactor operation.

If a leak occurs, the drywell will contain the released matter that will be present in the liquid, gaseous, and vapor phases. This will result in the collection of water in the sumps, a possible increase in drywell temperature, pressure, and relative humidity, an increase in the air-conditioning heat load, and an increase in the radioactivity of the drywell atmosphere. The closed limited volume of the drywell enhances the detection sensitivity.

There are 6 different methods used to detect leakage in the primary containment. These are outlined in Section 3.6.C discussion of this submittal. The different drywell parameters provide diverse methods for determining if an increased leak rate exists within the drywell. The allowable leakage rates have been based on the predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various drywell monitors.

Based on the behavior of cracks, a 5 gpm leak rate limit has been assigned to UNIDENTIFIED LEAKAGE and 20 gpm to IDENTIFIED LEAKAGE totaling a 25 gpm TOTAL LEAKAGE. Experience has shown that normal leak rate is 4 gpm into the equipment drain sump and 0 to 0.5 gpm into the floor drain sump.

The sump working capacities and sump discharge capacities are large enough to accept the design leak rates. The sump working capacity is the amount of water between the low level pump trip and the high-high-level alarm point. The equipment drain sump (approximate working capacity, 450 gallons) and the floor drain sump (approximate working capacity, 225 gallons) are drained by two 50 gpm pumps. This pump capacity permits one pump in each sump to remove the design total leakage because of the possibility that most of the leakage could flow into one sump.

The criterion for establishing the total leakage rate limit is based on the makeup capability of the CRD and RCIC systems and is independent of the feedwater system, normal ac power, and the emergency core cooling systems. The CRD system supplies 42 gpm into the reactor vessel; the RCIC system can supply 425 gpm through the feedwater sparger to the reactor vessel. The total leakage rate limit is set at less than 0.1 of this value or 25 gpm.

The proposed changes revise a shutdown requirement when the Reactor Coolant Leakage exceeds the LCO limits, by allowing a 4 hour period when the leakage can be brought back within limits. If the leakage cannot be reduced within the required time, a shutdown requirement to HOT SHUTDOWN and eventually COLD SHUTDOWN is initiated. The 4 hour period is a justified and accepted time frame by the NRC. The 4 hours is adequate time to allow the reduction and bring leakage into compliance with the limits specified in the TS. The probability of an accident exceeding the safety analysis during this 4 hour period is minimal and therefore does not increase the consequences of an accident. The existing TS require the reactor be placed in COLD SHUTDOWN when the leakage limits are exceeded. The proposed shutdown requirement is consistent with the other shutdown requirements within the DAEC TS and the guidance provided by the Standard TS.

The addition of the two LCOs is to provide specific guidance in the event the Sump System and/or Air Monitoring Systems are inoperable. The proposed changes are consistent with current plant practices, however, the wording of the existing LCOs is somewhat confusing. Applying human factors concepts, the existing LCOs have been revised to provide clarity and avoid potential confusion. The SRs have been revised to incorporate the changes made to the associated LCOs. These proposed changes do not affect the assumptions utilized to support the plant safety analysis or change any mitigation factors which have been credited in the DAEC licensing bases.

Other proposed changes, basically editorial, are also being made to provide consistency and clarity within the DAEC TS and to utilize the guidance provided by the Standard TS. Consequently, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

SAFETY AND RELIEF VALVES

The nuclear system pressure relief system includes two safety and six safety/relief valves located on the main steam lines within the drywell between the reactor vessel and the first isolation valve. The safety valves provide protection against the overpressure of the nuclear system and discharge directly to the interior space of the drywell. The safety/relief valves, which discharge to the suppression pool, provide the following three main functions:

1. Overpressure relief operation. The valves are opened to limit the pressure rise and prevent spring safety valve opening.

2. Overpressure safety operation. The valves augment the spring safety valves by opening in order to prevent nuclear system overpressurization.

3. Depressurization operation. The required valves are opened automatically or manually by indirectly operated devices as part of the protection system for small line breaks.

The main steam lines, in which the safety/relief and safety valves are installed, are designed, installed and tested in accordance with the applicable codes as discussed in the DAEC UFSAR Section 3.2.

The operational objective of the nuclear system pressure relief system is to prevent the opening of the spring-loaded safety valves during normal plant isolations and load rejections. The safety design bases are as follows:

1. The nuclear system pressure relief system prevents the overpressurization of the nuclear system to prevent the failure of the nuclear system process barrier because of pressure.

2. The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system occurring with maloperation of the HPCI system so that the LPCI and the Core Spray systems operate to protect the fuel barrier.

3. The safety/relief valve discharge piping is designed to accommodate forces resulting from relief action and is supported for reactions due to flow at maximum relief valve discharge capacity so that system integrity is maintained.

4. The nuclear system pressure relief system is designed for testing prior to nuclear system operation and for periodic verification of the operability of the nuclear system pressure relief system.

During power generation, design bases are as follows:

1. The nuclear system safety/relief valves prevent the opening of the spring-loaded safety valves during normal plant isolations and load rejections.

2. The nuclear system safety/relief valves discharge to the suppression pool below the water level to condense the exhaust steam.

3. The safety/relief valves will properly reclose following a plant

isolation or load rejection so that normal operations can be resumed as soon as possible.

The ASME Code requires overpressurization protection for vessels designed to meet Code Section III. The code permits a peak allowable pressure of 110% of vessel design pressure (1375 psig for a 1250 psig vessel). The code specifications for safety valves additionally require that the lowest safety valve setpoint be at or below vessel design pressure (1250 psig) and the highest valve setpoint be at or below 105% of vessel design pressure (1313 psig). The safety/relief valves are set to open by self-actuation (overpressure safety function) in the range from 1110 to 1140 psig, and the safety valves are set to operate at 1240 psig. These settings satisfy the ASME Code specifications for the setpoints of the safety valves.

For DAEC, the transient produced by the closure of all main steam line isolation valves represents the most severe abnormal operational transient resulting in a nuclear system pressure rise when direct scrams are ignored. The plant is assumed to be operating at the turbine generator conditions at a maximum vessel dome pressure of 1025 psig. The analysis hypothetically assumed the failure of the direct isolation valve position scram. The reactor is shut down by the backup, indirect, high-neutron-flux scram. For the analysis, the self-actuated setpoints (safety function) of the safety/relief valves are assumed to be in the range from 1121 to 1151 psig, and the safety valves are assumed to operate at 1252 psig (setpoint +1%). The safety/relief and safety valves open to limit the nuclear system pressure rise to 1275 psig. The analysis indicates that the design valve capacities are capable of maintaining adequate margin (100 psi) below the peak ASME Code allowable pressure in the nuclear system (1375 psig). The safety valve capacity in conjunction with safety/relief valve capacity limits the peak nuclear system pressure at the bottom of the vessel. The resulting criterion to the ASME Code limit ensures adequate protection against excessive overpressurization for the nuclear system process barrier even for this hypothetical reactor isolation event.

The analysis that forms the basis for the evaluation of the pressure relief function of the nuclear pressure relief system appears in the DAEC UFSAR Chapter 15. In summary, the opening of a relief valve or safety valve allows steam to be discharged into the primary containment. The sudden increase in the rate of steam flow reaching the reactor vessel causes the reactor vessel coolant inventory to decrease. The result is a mild depressurization transient.

The small amounts of radioactivity discharged with the steam are contained inside the primary containment; the situation is not significantly different, from a radiological viewpoint, than that encountered in cooling the plant using the relief valves to remove decay heat. This transient is a nonlimiting event (DAEC UFSAR reference 2 of Section 15.0); accordingly, only the foregoing narrative description of the event is provided.

As seen by the above discussion, none of the proposed changes deviate from the current safety analysis. The existing LCOs have been revised to include clear, concise, and specific shutdown requirements. In addition, a footnote has been added to provide a reference that some of the relief valves also perform ADS functions. These proposed changes provide consistency within the DAEC TS and utilize the guidance provided in the Standard TS. In addition, human factors concepts have been applied to these LCOs in order to minimize potential confusion.

The existing SR, requiring that 1 safety and 3 relief valves be checked per OPERATING CYCLE, has been revised incorporating the wording provided by the Standard TS. The proposed SR provides specific and detailed requirements ensuring that the safety and relief valves are tested in accordance with manufacturer's recommendations. The existing SR does not contain this level of detail. The proposed changes are an enhancement. These changes, along with the editorial changes, provide additional guidance not specifically provided in the existing TS. The proposed changes do not change any parameters or calculations used in the current safety analysis.

These changes provide additional assurance that the safety and relief valves will perform their intended function as described above.

JET PUMPS

DAEC has two external recirculation loops each discharging high-pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the driven flow for the jet pumps. This flow enters the jet pumps at the suction inlet and is accelerated by the driving flow. The driving and driven flows are mixed in the jet pump throat section resulting in partial pressure recovery. The balance of recovery is obtained in the jet pump diffusing section as referenced in DAEC UFSAR Section 5.4-5. The adequacy of the total flow to the core is discussed in DAEC UFSAR Section 4.4. Jet pump operating experience has shown that the design is sound and that the jet pump operation is stable and predictable.

From a safety analysis standpoint, there is no specific jet pump analysis.

Any jet pump event is enveloped into the safety analysis of the recirculation system. The only other event involving the jet pumps is the cracking of the hold down beams. This issue was brought before the industry in the NRC issued Bulletin 80-07. This bulletin required, in addition to performing examinations of these hold down beams, that a surveillance program to monitor jet pump performance be initiated and continued until plant TS could be changed. Iowa Electric initiated and performed the required monitoring until NRC approved and issued TS Amendment 158, dated April 28, 1989. This approved Amendment was developed using the guidance provided by General Electric issued SIL No. 33, "Jet Pump Beam Cracks." This SIL discusses the jet pump beam failure problem and provides recommendations for modifications to the TS in order to improve detection of any impending failure of these beams.

The proposed changes to this section do not change any of the surveillance requirements approved by the NRC with regard to this concern.

The existing TS requires that if an engineering evaluation determines that a jet pump is inoperable, the reactor must be brought to COLD SHUTDOWN. The proposed TS would require the reactor to be brought to HOT SHUTDOWN instead of COLD SHUTDOWN. The intent of the TS action/shutdown statements is that if a system is inoperable and a MODE change is required, that the reactor be placed in a MODE to which the LCO does not apply. The TS require jet pumps to be OPERABLE in the RUN and STARTUP MODES. Therefore, HOT SHUTDOWN would be the first MODE in which the Jet Pump is not required to be OPERABLE. Making this change eliminates the requirement to cycle the plant to colder temperature conditions than what is actually needed, thus not adding any unnecessary thermal stresses to the system. This proposed change does not alter the safety analysis, calculations, or parameters used to support the licensing bases of DAEC.

In addition, several editorial changes are being proposed to provide additional clarification and consistency with current plant practices. These changes do not change the intent of the existing TS and therefore, do not involve a significant increase in the probability or consequences of an accident previously evaluated.

<u>JET PUMP FLOW MISMATCH</u>
Reference the above section, "Jet Pumps" for additional discussion on safety analysis.

The existing TS requires that if recirculation pump speed is outside its limits, the pump must be tripped and single loop operation (SLO) entered. The proposed TS allows 2 hours to bring the recirculation pump back to within its limits prior to initiating SLO. The 2 hours is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the action, and on the frequent core monitoring by operators allowing abrupt changes in core flow condition to be quickly detected. In addition, DAEC has been analyzed for SLO in NEDO-24272 dated July 1980, with SLO approved by TS Amendment 119 dated May 1985. The proposed change is more conservative and is encompassed within the SLO envelope. This change does not result in deviating or departing from any existing calculations, safety analysis, or accident mitigation actions, and is consistent with the guidance provided by the Standard TS.

Editorial changes have been proposed to provide clarity, consistency, and additional guidance not included in the existing TS.

STRUCTURAL INTEGRITY

Three LCOs were added to this TS stating the thermal and pressurization limits and actions to be taken if those limits for any ASME Section XI Class 1, 2, or 3 component(s) are exceeded. In the event one of the subject component(s) cannot be restored to within these limits, an action statement allows isolation of the component(s) with specific restrictions stated in each separate LCO. The existing TS do not contain these LCOs. This change will enhance the existing TS and ensure that the Class 1, 2, and 3 component(s) perform in accordance with the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50.55a(g), except where specific written relief has been granted by the NRC. The proposed changes do not alter programs previously approved by the NRC, calculations, nor safety analysis utilized by DAEC. The proposed action statements are consistent with the guidance provided in the Standard TS.

The two SRs being deleted are not actually SRs. They contain information which is located in the Bases. The deletion of these SRs will not eliminate any testing or verification of system OPERABILITY requirements or actions. These proposed changes do not degrade the intent of the structural integrity TS, any safety related components, or alter any intended safety functions of any equipment.

The editorial changes are intended to enhance the TS by providing clarity and consistency. The guidance provided by the Standard TS was utilized in making these proposed changes. These changes comply with the guidance and examples provided in 51FR7751 dated March 6, 1986, that are considered not likely to involve a significant hazards consideration.

SHOCK SUPPRESSORS (SNUBBERS)
As determined by the Staff, the alternate schedule for visual inspections proposed in Table 4.6.H-1 provides the same level of confidence as the existing schedule. The actions required by the existing TS as a result of finding snubbers inoperable remain the same.

Several LCOs and SRs have been revised. The intent of these revisions is to provide clarity and consistency throughout the TS section. These changes are editorial in nature as discussed in 51FR7751 dated March 6, 1986. These changes utilize the guidance provided in the Standard TS for the shutdown requirements. These changes do not alter DAEC TS for safety functions, plant equipment, or calculations in the safety analysis.

This proposed Amendment also adds SRs for Transient Event Inspections, Functional Testing of Repaired and Replacement Snubbers, and Snubber Service Life Replacement Program. The existing TS do not contain these subject SRs. The proposed SRs will provide assurance that the snubbers perform their intended function. The addition of these SRs will also ensure that the snubbers will assist the associated supported system in performing its intended function as required in the DAEC FSAR.

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes discussed in this section are provided to enhance the overall quality and safety significance of the existing DAEC TS. The proposed TS do not change any accident analysis, plant safety analysis, calculations, degrade existing plant programs, modify any functions of safety related systems, or accident mitigation functions DAEC has previously been credited with. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

The proposed changes to the Bases Section 3.6 and 4.6 reflect the above changes and include various editorial corrections. These changes have no effect on the consequences of a previously evaluated accident.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not alter any plant parameters, revise any safety limit setpoint, or provide any new release pathways. In addition, the proposed changes do not modify the operation or function of any safety related equipment, nor do they introduce any new modes of operation, failure modes, or physical changes to the plant. The proposed changes do not change any plant parameters or transient responses assumed in the Design Bases of the plant and therefore, do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the Bases Section 3.6 and 4.6 reflect the above changes and include various editorial corrections. Therefore the proposed changes and corrections do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in the margin of safety.

The proposed changes do not require any modifications to existing plant systems or equipment, Emergency Operating Procedures, safety limit settings, or parameters utilized in the licensing bases for the safety analysis. These proposed changes are being made to enhance TS Section 3.6 by clarifying and making LCOs and SRs consistent throughout the section. In addition, several LCOs and SRs have been added, providing additional information that did not exist in the current TS. As discussed above, the proposed changes do not change any safety analysis or any accident mitigation actions for which DAEC has previously taken credit. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed changes to the Bases Section 3.6 and 4.6 reflect the above changes and include various editorial corrections. These changes do not involve a significant reduction in the margin of safety.

In conclusion, the Commission has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (March 6, 1986, 51FR7751) of amendments that are considered not likely to involve a significant hazards consideration. Although the majority of the proposed changes are directly comparable to the examples, several other proposed changes are not; however, as stated above, the changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.