PROPOSED CHANGE RTS-197A TO THE DUANE ARNOLD ENERGY CENTER TECHNICAL SPECIFICATIONS

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

List of Affected Pages

ii*
vi**
1.0-10
1.0-11***
3.6-1 through 3.6-41*

*This page is common to RTS-249 (as well as pages 3.6-9, 3.6-11, 3.6-12, 3.6-13, 3.6-24, 3.6-29).
**This page is common to RTS-246.
***This page reflects changes made by Amendment 193 (RTS-186).

Summary of Changes:

The following list of proposed changes is in the order that the changes appear in the Technical Specifications. The specific page number, listed in the left margin, represents the existing TS page number. For those pages that have been added or if the revised TS falls on a different page then in the existing TS, no page number will be specified in the left margin.

PAGE	**DESCRIPTION OF CHANGES**
ii	The Table of Contents has been revised to reflect the changes in Section 3.6, "Primary System Boundary."
vi	The List of Tables has been revised to reflect the addition of three Tables in Section 3.6.

DAEC TS Section 1.0, DEFINITIONS

The following are proposed changes to the DAEC Definitions Section of the TS.

1.0-10 Revise Existing Page 1.0-10 to 1.0-11

Amendment 193 (RTS 186) has been approved by the NRC, but has not yet been implemented in the DAEC TS. In the subject RTS, Table 1.0-1, "OPERATING MODES" was introduced as page 1.0-10. In this proposed TS, RTS 197, the definitions discussed below are being proposed. In order to maintain consistency, DAEC is adding these definitions before the Table. This will make the Table page 1.0-11 and the proposed definitions page 1.0-10. This is strictly an editorial change.

Add Definition 41, IDENTIFIED LEAKAGE

Add the following Definition:
"IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are

both specifically located and known not to interfere with the operation of the leakage detection systems."

The existing DAEC TS do not define IDENTIFIED LEAKAGE. This definition was added to provide clarity and consistency with LCO 3.6.C and the Standard TS definition.

Add Definition 42, TOTAL LEAKAGE

Add the following definition:
"TOTAL LEAKAGE is the sum of IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE."

The existing DAEC TS do not define TOTAL LEAKAGE. This definition is added to provide clarity and consistency with LCO 3.6.C and the Standard TS.

Add Definition 43, UNIDENTIFIED LEAKAGE

Add the following definition:
"UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE."

The existing DAEC TS do not define UNIDENTIFIED LEAKAGE. This definition was added to provide clarity and consistency with LCO 3.6.C and the Standard TS.

Add Definition 44, DOSE EQUIVALENT I-131

Add the following definition:
"DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram(ml), which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

The existing DAEC TS do not contain a definition of DOSE EQUIVALENT I-131. This definition was added to provide clarity and consistency with LCO 3.6.B and the Standard TS.

DAEC TS Section 3/4.6.A, Thermal and Pressurization Limitations

The following are proposed changes to the DAEC Thermal and Pressurization Limitations Section of the TS.

3.6-1 Revise existing 3/4.6 Applicability Statement

Revise the existing Applicability Statement as follows:
"Applies to the operating status of the **R**eactor **C**oolant **S**ystem."

This is an editorial change which uses initial caps for "Reactor Coolant System." For consistency, DAEC TS Section 3.6 is being revised to use initial caps on all system names when the word system proceeds the specific name.

3.6-1 Revise existing Objective Statement

Revise the existing Objective Statement as follows:
"To assure the integrity and safe operation of the **R**eactor **C**oolant **S**ystem."

This is an editorial change which uses initial caps for "Reactor Coolant System." For consistency, DAEC TS Section 3.6 is being revised to use initial caps on all system names when the word system proceeds the specific name.

3.6-1 Revise existing LCO 3.6.A.1

The existing LCO is being revised as follows:
"The average rate of reactor coolant temperature change during normal heatup ~~or~~ **and** cooldown shall not exceed 100°F/hr when averaged over a one-hour period."

The word "or" is being replaced with the word "and". This makes the wording consistent with SR 4.6.A.1.

3.6-1 Revise existing LCO 3.6.A.2

The existing LCO is being revised as follows:
"The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or

greater than that shown in Curve C ~~or~~ **of** Figure 3.6-1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Figure 3.6-1. ~~Figure 3.6-1 is effective through 16 effective full power years. At least six months prior to 16 effective full power years new curves will be submitted.~~"

The word "or" is being replaced with the word "of". This corrects a typographical error.

This LCO currently contains a statement, "Figure 3.6-1 is effective through 16 effective full power years. At least six months prior to 16 effective full power years new curves will be submitted."

This statement is being deleted for the following reasons:

a) This statement is a commitment and not an LCO. This statement does not contain information relative to the immediate operation of the plant or the mitigation of any accident. Information of this nature is typically located in the Bases Section and/or in a commitment tracking system. DAEC is proposing to revise the Bases Section to contain this information.

b) NRC Generic Letter (GL) 91-01, "REMOVAL OF THE SCHEDULE FOR THE WITHDRAWAL OF REACTOR VESSEL MATERIAL SPECIMENS FROM TECHNICAL SPECIFICATIONS" provides guidance for the removal of the specimen schedule and associated language from the TS. The justification for removal is that the TS requirement is a duplication of 10CFR50, Appendix H. The GL also provides that the licensee state in the TS that the specimens be withdrawn in accordance with 10CFR50, Appendix H. The proposed DAEC SR 4.6.A.2 has been revised to comply with the GL. As stated in the GL, the removal of the schedule and language associated with the schedule will not result in any loss of regulatory control.

In general, 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. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. 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 are 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 effects 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 1 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 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 or mitigation of any accident and does not affect any procedural steps in the Emergency Operating Procedures. The proposed revision will not result in any loss of regulatory control since DAEC is still maintaining the requirements specified in 10CFR50, Appendix H.

3.6-2 <u>Revise existing LCO 3.6.A.3</u>

The existing LCO is being revised as follows:
"The reactor vessel head bolting studs shall not be under tension unless the temperature of the ~~vessel~~ head flange and the **shell adjacent to the** head **flange** is greater than **or equal to** 74°F."

This proposed revision provides additional clarification and

guidance. The word "vessel" is being deleted to be consistent with the proposed SR 4.6.A.3. The word "vessel" is superfluous and does not add any clarification to the specification. In addition, the words "**shell adjacent to the** head **flange**" are being added to identify the specific location where the temperature has been and will continue to be taken.

The current TS state that "... the temperature of the vessel head flange and the head is greater than 74°F." This LCO has been revised to state,"... the temperature of the head flange and the **shell adjacent to the** head **flange** is greater than **or equal to** 74°F." The Bases currently state that the closure flange region, with RT_{NDT} = 14°F, has a bolt preload and minimum operation temperature of 74°F.

The minimum temperature for boltup and pressurization of 74°F was established by adding 60°F to the RT_{NDT} for the limiting closure flange region. The 60°F added to the RT_{NDT} for boltup and pressurization is a requirement of the ASME Code applicable to the original reactor pressure vessel design work. However, Appendix G of the 1984 ASME Code, Section III, through the Summer 1984 Addenda requires a minimum permissible temperature of RT_{NDT} for boltup and pressurization up to 20% of hydrotest pressure. The 60°F added to the RT_{NDT} is extra margin included because the closure flange region stress analysis assumes a 0.24 inch flaw (which is detectable) instead of a 1/4 inch T flaw. In the case of the core critical operation curves (C and C') in UFSAR Figure 5.3-1, 10CFR50, Appendix G, Paragraph IV.A.3 requires a minimum permissible temperature of (RT_{NDT} = 14°F + 60°F) 74°F.

The minimum temperature for boltup prior to pressurization must be 74°F or greater. Boltup at 74°F satisfies the requirements of the original code of construction and exceeds the Summer 1984 Addenda code requirements. A sufficient number of studs may be partially tensioned to seal the closure flange O-rings for the purpose of raising reactor water level above the closure flanges, in order to assist in warming the flanges and adjacent shell to a minimum temperature of 74°F before they are stressed by the full intended bolt preload.

The existing TS state that the reactor vessel head bolting studs shall not be tensioned unless the temperature is "greater than 74°F". Thus, there is an inconsistency between the existing Bases 3.6.A and 4.6.A and existing TS LCO 3.6.A.3. The Bases would allow the bolts to be tensioned at 74°F or greater whereas the TS would

not. Therefore, adding the words **"or equal to"** brings the TS and Bases in agreement.

Add new LCO 3.6.A.4

Add the following LCO:
"With any of the above limits exceeded:

a. restore the temperature and/or pressure to within the limits within 30 minutes, and

b. within 72 hours perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System, and

c. determine that the Reactor Coolant System remains acceptable for continued operation; or

d. if the above requirements cannot be met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours."

The existing TS does not specify action to be taken in the event that the temperature/pressure limits are exceeded. The proposed action statement specifies the action. Instead of immediately initiating a shutdown, the operators would have 30 minutes to restore the temperature/pressure limits prior to initiation of a controlled shutdown. This period has been accepted by the NRC as sufficient time to restore the temperature/pressure limits; it also reflects the urgency of restoring the parameters to within the analyzed range. In addition, the proposed action statement requires performing an engineering evaluation within 72 hours after bringing the reactor temperature and pressure limits back within limits. This evaluation is to determine the effects of the out-of-limit condition on the structural integrity of the RCS and to determine that it remains acceptable for continued operation. If the temperature/pressure conditions cannot be brought back into limits within the 30 minute interval, or the engineering evaluation does not support continued operation, the reactor is to be in HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours. These intervals are based on previous plant shutdown history within the industry and comply with the guidance provided by Standard TS. This proposed change is an enhancement to the current TS. It provides specific guidance to the operators on

how to remedy the out-of-limit condition and what actions to take if the out-of-limit condition cannot be remedied.

Add new LCO 3.6.A.5

Add the following LCO:
"When in RUN, STARTUP, HOT SHUTDOWN, COLD SHUTDOWN, or REFUELING MODE, the following LCOs apply to the idle recirculation loop startup:"

The current TS identifies specific requirements that the recirculation pumps must meet; however, it does not specifically identify what MODES or operational conditions the recirculation pumps need to be in when meeting these requirements. DAEC has adopted the Standard TS language and guidance to better define the MODES or operational conditions in which these requirements actually apply. This LCO was added to specifically identify that the recirculation pumps must comply with the LCO when the reactor is in the RUN, STARTUP, HOT SHUTDOWN, COLD SHUTDOWN, or REFUELING MODES.

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 transients 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 of 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).

With no recirculation flow, the coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel. 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.

This proposed change provides additional clarification and guidance to the operator as to the applicable MODES or operational conditions when starting a recirculation pump. It can also be seen that the proposed changes do not exceed any temperature limits as analyzed in the DAEC UFSAR.

3.6-2 Revise existing LCO 3.6.A.5 to new LCO 3.6.A.5.a

The existing LCO is being revised as follows:
"~~The~~ **A** reactor recirculation pump~~s~~ shall not be started unless the **reactor** coolant temperature~~s~~ **differential** between the dome and the bottom head drain ~~are within~~ **is less than or equal to** 145°F."

The word "The" has been replaced by "**A**". This is an editorial change only.

The words "**reactor**" and "**differential**" are added to the above LCO for clarification. In addition, the words "pumps" and "temperatures" have been made singular to make the sentence grammatically correct.

The existing TS states, "... the coolant temperatures between the dome and the bottom head drain are within 145°F." This has been revised. The words "are within" have been replaced with the words, "...**is less than or equal to** 145°F." This revision makes the TS consistent with the Bases. This proposed change makes the wording consistent throughout Section 3.6 and does not change the intent of the LCO. This proposed change clarifies that the temperature range includes 145°F, which is within the limits of the DAEC Safety Analysis. The Bases state that, "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 which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F."

3.6-2 Revise existing LCO 3.6.A.4 to new LCO 3.6.A.5.b

The existing LCO is being revised as follows:
"~~The pump in an idle~~ **When only one** recirculation loop **is idle, the recirculation pump** shall not be started unless the temperature~~s~~ **differential** of the **reactor** coolant ~~within~~ **between** the idle and operating recirculation loops ~~are~~ **is less than or equal to** ~~within~~ 50°F. ~~of each other.~~"

The words "The pump in an idle" are being replaced with the words "**When only one.**" This is an editorial change. The intent of this LCO is that the idle recirculation pump shall not be started unless the above temperature conditions are met. This change provides clarity and is also consistent with the guidance provided by the Standard TS.

The words "... **is idle, the recirculation pump** ... ." are being added to the existing LCO for clarification. This statement is to clarify and specifically state the conditions of an idle recirculation loop prior to being started.

The word "temperature" has been made singular thus providing the proper tense to grammatically make the sentence correct based on the proposed changes.

The word "**differential**" is being added for clarity. The intent of the TS is to measure the temperature differential between the idle loop and the operating loop.

The word "**reactor**" was added preceding the word "coolant" for clarification. This is an editorial change being made for consistency throughout the DAEC TS Section 3.6. The intent of the existing LCO has not changed based on this editorial change.

The word "within" is being replaced with "**between.**" The LCO discusses the temperature differential of the reactor coolant which is to be measured in the idle loop and the operating loop. This differential is to be within 50°F before starting the idle pump. The current TS state that the coolant temperature in the idle and operating loops is within 50°F. It is more accurate to refer to the reactor coolant temperature differential **between** the idle and operating loops. This change does not change the intent of the existing TS.

The words "**is less than or equal to**" are being added. This change clarifies the existing LCO ensuring that the temperature range includes the 50°F. The existing TS imply this, however, the

proposed change ensures the clarity of the intent. This wording is also consistent with the wording being proposed in other LCOs within this section of the DAEC TS.

The words "are within ... of each other." are being deleted as a result of adding the word "**between**" as discussed above and to make the sentence read correctly based on the additional revision to this LCO.

These changes discussed above are for the most part editorial. The proposed changes enhance clarity, provide consistency, and bring the LCO into closer compliance with the guidance provided by Standard TS.

3.6-1 Revise the existing 4.6 Applicability Statement

The existing Applicability statement is being revised as follows: "Applies to the periodic examination and testing requirements for the ~~r~~ **R**eactor ~~c~~ **C**oolant ~~s~~ **S**ystem."

This is an editorial change which uses initial caps for "Reactor Coolant System." For consistency, the DAEC TS Section 3.6 is being revised to use initial caps on all system names when the word system proceeds the specific name.

3.6-1 Revise the existing 4.6 Objective Statement

The existing Objective statement is being revised as follows: "To determine the condition of the ~~r~~ **R**eactor ~~c~~ **C**oolant ~~s~~ **S**ystem and the operation of the safety devices related to it."

This is an editorial change which uses initial caps for "Reactor Coolant System." For consistency, the DAEC TS Section 3.6 is being revised to use initial caps on all system names when the word system proceeds the specific name.

3.6-1 Revise existing SR 4.6.A.1

The existing SR is being revised as follows:
"During heatups and cooldowns, the ~~following~~ temperatures **at the following locations** shall be ~~logged~~ **recorded** at least every 15 minutes until 3 consecutive readings at each ~~given~~ location are

within 5°F:

a. ~~R~~ reactor vessel shell adjacent to shell flange.

b. ~~R~~ reactor vessel bottom drain.

c. ~~R~~ recirculation loops A and B.

d. ~~R~~ reactor vessel bottom head temperature."

This SR has been revised with several editorial changes providing clarity and consistency. The existing SR states, "... the ~~following~~ temperatures shall be ... ." This SR has been revised to state, "... temperatures **at the following locations** shall be ... ." The proposed change is more accurate, grammatically correct, and does not change the intent of the existing requirements.

The word "logged" has been replaced with **"recorded."** The word "logged" implies that a reading must be written down by an individual; the change more clearly permits use of a recorder/computer for monitoring and recording temperatures.

The word "given" has been deleted because it is a superfluous word and does not add anything to the SR. This change is editorial in nature.

Changing all the initial letters in SR 4.6.A.1.a through d from initial caps to lower case letters is editorial.

3.6-1/2 <u>Revise existing SR 4.6.A.2</u>

The existing SR is being revised as follows:
The first paragraph has not been changed. The existing second and third paragraphs of this SR are being deleted.

Add the following paragraph:
"The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER as required by 10CFR50, Appendix H. The results of these fluence determinations shall be used to update Figure 3.6-1."

The paragraphs deleted discuss how the test specimens of the reactor base were installed and when and how the surveillance capsules are

withdrawn. This information has been incorporated into the bases of the TS.

The added paragraph simply states that the specimens will be removed and examined in accordance with IOCFR50, Appendix H and that the results of the fluence determinations will be used to update Figure 3.6-1, "Pressure Limit in Reactor Vessel Top Head versus Minimum Reactor Vessel Metal Temperature." This proposal is consistent with the NRC Generic Letter 91-01 and the guidance provided by Standard TS.

As required by 10CFR50, Appendix H, specimens of reactor vessel material are installed near the inside reactor vessel wall and are withdrawn on a schedule to provide data as to the effect of radiation and thermal environment on the vessel material. Using this data, the temperature/pressure limits curve (TS Figure 3.6-1) is adjusted, as necessary, to compensate for the shift in material transition temperature as indicated by tests of the withdrawal specimens. This provides assurance that DAEC is operating in the ductility region of the vessel material. Exposure to a potential for brittle fracture is then precluded.

Section II.B.3 of Appendix H requires that the specimen withdrawal schedule be submitted to and approved by the NRC prior to program implementation. As noted in the reference, having the withdrawal schedule in the TS duplicates the controls established by Appendix H. In addition, the referenced Generic Letter also required a commitment to include the withdrawal schedule in the next revision of the Safety Analysis Report and to maintain it in the Safety Analysis Report. At the next FSAR Update, the details of the removal schedule will be incorporated.

This revision provides clarity and removes some information to the Bases Section.

3.6-2 <u>Existing SR 4.6.A.3</u>

There were no changes made to this section.

<u>Add new 4.6.A.4</u>

Add the following SR:
"Prior to starting a recirculation pump, the following reactor

coolant temperatures shall be within limits and recorded:"

The existing SRs 4.6.A.4 and 4.6.A.5 have been revised, reorganized, and incorporated into the proposed 4.6.A.4, 4.6.A.4.a, and 4.6.A.4.b. The proposed revision divides the existing subject SRs into two itemized SRs.

The proposed SR 4.6.A.4 has been added as the introduction section supporting SRs 4.6.A.4.a and 4.6.A.4.b. In addition, this change provides clarity and is consistent with the itemized format used by Standard TS and throughout Section 3.6. The proposed change does not modify the intent of the existing LCO.

3.6-2 Revise existing SR 4.6.A.4 to SR 4.6.A.4.b

The existing SR is being revised as follows:
"~~Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.~~" to "**differential between the recirculation loops.**"

This SR has been revised by deleting most of the sentence since it is incorporated and consistent with the proposed SR 4.6.A.4. This revision is editorial in nature and provides clarity in specifically itemizing the SR. This SR will ensure that the requirements specified in LCO 3.6.A.5.b are complied with. This revised SR does not delete or deviate from any requirements as specified in the existing SR.

3.6-2 Revise existing SR 4.6.A.5 to SR 4.6.A.4.a

The existing SR is being revised as follows:
"~~Prior to starting a recirculation pump, the reactor coolant temperatures in~~ **differential between** the dome and ~~in~~ the bottom head drain ~~shall be compared and permanently logged~~."

This SR has been revised by deleting most of the sentence since it is incorporated and consistent with SR 4.6.A.4. This revision is editorial in nature and provides clarity in specifically itemizing the SR. This SR will ensure that the requirements specified in LCO 3.6.A.5.a are complied with. This revised SR does not delete or deviate from any required SR requirements as specified in the existing SR.

DAEC TS Section 3/4.6.B, COOLANT CHEMISTRY

Major revisions are being made in the format of this section. Much of the existing information has been retained but has either been placed in Table format or in shorter LCO and SR statements grouped by reactor MODE. This will make the requirements more understandable than in their current format. The existing TS were written when DAEC used gross iodine radiochemistry methodology and sodium iodide detectors. Currently, DAEC is using the Dose Equivalent Methodology and no longer uses sodium iodide detectors. The proposed changes reflect the use of the Dose Equivalent Methodology. In addition, the units of measurement have been changed to reflect those used at DAEC. Instead of using grams and ppm, DAEC uses ml (milliliter) and ppb. The presentation of this section will differ from the other sections in this amendment. For simplicity, the existing LCOs and SRs are being deleted with the proposed changes appearing in their entirety on the separate proceeding pages.

3.6-3 Revise existing LCO 3.6.B.1

3.6-3 Revise the existing LCO as follows:
"~~Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 1.2 μCi/gm of dose equivalent* I-131. This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient, the iodine concentrations shall not exceed the equilibrium values by a factor of more than 10 whenever the reactor is critical. The reactor shall not be operated more than 5 percent of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds the equilibrium limit by a factor greater than 10, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.~~

~~*That concentration of I-131 which alone would produce the same thyroid dose as the quantity and isotopic mixture actually present.~~"

The existing LCO is being divided into three different LCOs. LCO 3.6.B.1.a, 3.6.B.1.b, and 3.6.B.1.c.

Proposed LCO 3.6.B.1.a states, "**With the reactor critical, the specific activity of the primary coolant shall be less than or equal to 1.2 μCi/ml DOSE EQUIVALENT I-131.**" This replaces the first sentence of the existing LCO. There have been minor editorial changes made in order for this LCO to stand alone. These minor

changes do not change the intent of the existing LCO.

Proposed LCO 3.6.B.1.b states, **"When in the RUN, STARTUP, or HOT SHUTDOWN MODES, the specific activity of the primary coolant can be greater than 1.2 μCi/ml DOSE EQUIVALENT I-131 for a maximum of 48 hours, provided that the DOSE EQUIVALENT I-131 activity does not exceed 12.0 μCi/ml during this time. The reactor shall not be operated more than 5 percent of its yearly power operation under this exception for equilibrium activity limits."** This LCO is replacing the requirements contained in the second, third and fourth sentences of the existing LCO. There are minor editorial changes made in order for this LCO to stand alone.

The statement, **"When in the RUN, STARTUP, or HOT SHUTDOWN MODES, ..."** is added to identify the reactor MODE to which the specific LCO applies. This is consistent with the rest of the DAEC TS Section 3.6.

The term **"DOSE EQUIVALENT"** appears in all caps since it is a defined term in the DAEC TS Section 3.6.

The existing TS state that, "the iodine concentrations shall not exceed the equilibrium values by a factor of more than 10 whenever the reactor is critical." This is incorporated into the proposed TS. However, instead of specifying a factor of 10, the proposed TS uses the actual value of 12.0 μCi/ml.

The rest of the existing LCO 3.6.B.1 is incorporated in proposed TS LCO 3.6.B.1.c which states, **"If the specific activity of the primary coolant is greater than 12.0 μCi/ml DOSE EQUIVALENT I-131, the reactor shall be shutdown, and the Main Steam Line Isolation Valves shall be closed immediately."** Only minor changes are proposed to make the LCO stand alone. The existing TS state that "If the iodine concentration in the coolant exceeds the equilibrium limit by a factor greater than 10, the reactor shall be shutdown." This has been replaced using the actual value of 12.0 μCi/ml. These changes do not change the intent of the existing LCO.

The footnote, "That concentration of I-131 which alone would produce the same thyroid dose as the quantity and isotopic mixture actually present." is no longer needed and will be deleted.

3.6-3b <u>Revise existing 3.6.B.2.a to Proposed LCO 3.6.B.2, 3.6.B.2.a, and 3.6.B.2.a.1</u>

Revise the existing LCO as follows:
"~~The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour, during SHUTDOWN or when in the REFUELING MODE:~~

~~Conductivity~~	~~5 μmho/cm~~
~~Chloride ion~~	~~0.1 ppm~~

~~At all times when the conductivity exceeds 5 micromhos/cm, the pH shall not be less than 4.6, except that short term spikes of up to two hours duration each are permissible in the pH range 4.0 to 4.5 and of up to four hours duration each, in the range 4.5 to 4.6. The total time in which the conductivity exceeds 5 micromhos/cm shall not exceed 720 hours."~~

Proposed LCO 3.6.B.2 states,
"At all times the chemistry of the Reactor Coolant System shall be maintained within the limits specified in Table 3.6.B.2-1."

This proposed change references the limits as specified in the new Table 3.6.B.2-1. Placing and referencing information in the Tabular format presents the information in a clear and concise manner.

Add new LCO 3.6.B.2.a

"In RUN MODE:"

The current specifications of this section have been divided into MODES to provide clarity and consistency.

Proposed LCO 3.6.B.2.a.1 states, **"With any limit in Table 3.6.B.2-1 exceeded for more than:**

a) 720 hours per year, or

b) 72 continuous hours,

be in at least STARTUP within 6 hours."

The existing LCO 3.6.B.2.a has been revised and split into specific MODES of applicability as provided by the guidance in the Standard TS. The limits in the existing TS have been relocated to Table 3.6.B.2-1. LCO 3.6.B.2.a.1 provides the time limits in which the chemistry can exceed the limit specified in Table 3.6.B.2-1 and the appropriate shutdown requirements when in the RUN MODE.

The existing LCO does not provide clear guidance concerning shutdown in the event that the 720 hours per year is exceeded. The proposed LCO provides that guidance by requiring the reactor to be in at least STARTUP within the next 6 hours. This is consistent with the guidance provided by the Standard TS.

3.6-4 Revise existing LCO 3.6.B.2.b to Proposed LCO 3.6.B.2.a.2

Revise the existing LCO as follows:
"~~The reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hour:~~

~~Conductivity 10 μmho/cm~~
~~>5 μmho/cm 2 weeks/year~~
~~Chloride ion 0.5 ppm~~
~~>0.2 ppm 2 weeks/year"~~

Existing LCO 3.6.B.2.b has been revised and split into specific MODES of applicability as provided by the guidance in the Standard TS. Table 3.6.B.2-1 incorporates the specific limits for chlorides, conductivity, and pH for all modes of operation. The associated action and shutdown statements are contained in the proposed LCO 3.6.B.2.a.2 which states, "**With the conductivity exceeding 10.0 μmho/cm at 25°C or chloride concentration exceeding 500 ppb, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.**" The existing TS did not contain a clear and concise shutdown requirement. This requirement is consistent with the guidance provided by the Standard TS.

Add new LCO 3.6.B.2.a.3

Add the following LCO:
"**Continuously record the conductivity of the reactor coolant. With no continuous recording conductivity monitor OPERABLE, install a temporary in-line conductivity monitor within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.**"

This LCO is being added to ensure that the continuous monitoring requirement is maintained. In addition, the LCO provides specific actions and shutdown requirements in the event that continuous monitoring is not maintained. The standard shutdown requirement,

"be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours." is being added. This change is being made throughout the entire section for consistency and clarity.

3.6-4 Delete existing LCO 3.6.B.2.c

Delete the existing LCO as follows:
"~~Every effort will be made to keep the conductivity below 1 μmho/cm at all times.~~"

This LCO is being deleted. It is always the intent to keep the conductivity as low as possible and in compliance with the limits shown in the proposed Table. This is not a LCO statement and should not be located here.

3.6-4 Revise existing LCO 3.6.B.2.d

Revise the existing LCO as follows:
"~~During power operation if the conductivity exceeds 1.0 μmho/cm, pH shall be measured and brought within the range of 5.6 to 8.6 within 24 hours. If the pH cannot be corrected, or if the pH remains outside the range of 4 to 10 the reactor coolant temperature shall be reduced to <212°F.~~"

The existing LCO has been divided into a Table and other LCOs. The pH values and MODES of applicability are now located in proposed Table 3.6.B.2-1. The present DAEC values are reflected in this Table. The existing LCO has been divided into proposed LCOs 3.6.B.2.b and 3.6.B.2.c (below) depending on the reactor MODE. The existing TS were not as clear as to the actions to take in any MODE of operation. The revised TS provides specific guidance as to the actions. This guidance is consistent with that provided in the Standard TS.

Add the proposed LCO 3.6.B.2.b

Add the following LCO:
"b. In STARTUP and HOT SHUTDOWN:

1) With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.B.2-1 for

more than 48 continuous hours, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours."

Add the proposed LCO 3.6.B.2.c

Add the following LCO:
"c. In COLD SHUTDOWN and REFUEL:

1) With the conductivity or pH exceeding the limit specified in Table 3.6.B.2-1, restore the conductivity and pH to within the limit within 72 hours.

2) With chloride concentration exceeding the limit specified in Table 3.6.B.2-1, restore the chloride concentration to within the limit within 24 hours."

3.6-4 Delete existing LCO 3.6.B.3.a

Delete the following LCO:
"~~Reactor coolant water shall not exceed the limit specified in 3.6.B.2 above.~~"

The substance of this LCO has been relocated into proposed TS 3.6.B.2.a.1, 3.6.B.2.a.2, 3.6.B.2.b.1, 3.6.B.2.c.1, and 3.6.B.2.c.2. The existing requirements for the pH have been separated into specific MODES of operation. Each one of these proposed LCOs contains shutdown requirements in the event that the pH is not within the limits specified in Table 3.6.B.2-1. The proposed TS provide clear and concise requirements and actions for the pH limits and shutdown requirements.

3.6-4 Delete existing LCO 3.6.B.3.b

Delete the following LCO:
"~~If one of the three monitors is inoperable for a period greater than 30 days, the plant will be shut down in an orderly manner.~~"

The intent of this LCO has been relocated to proposed LCO 3.6.B.2.a.3 and revised to incorporate the Standard TS requirements and shutdown statement. The Standard TS require that continuous monitoring be provided and does not specifically state how many

monitors are required or their specific locations. The current DAEC TS provide such detail which is not needed. DAEC TS are proposed to be revised requiring continuous monitoring and the standard shutdown statement in the event that continuous monitoring or a temporary in-line monitor cannot be installed within 4 hours. Specific locations of the monitors are located in the Bases.

3.6-4 Delete existing LCO 3.6.B.4

Delete the following LCO:
"~~If Specification 3.6.B is not met, an orderly shutdown shall be initiated.~~"

This LCO is being deleted since each proposed LCO contains its own shutdown requirement as applicable.

Add the proposed LCO 3.6.B.2.d

Add the following LCO:
"If the conditions in Specification 3.6.B.2.c.1 or 3.6.B.2.c.2 above cannot be met:

1) perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System, and

2) determine that the structural integrity of the Reactor Coolant System remains acceptable for continued operation prior to leaving COLD SHUTDOWN."

The existing TS do not require an engineering evaluation as stated above. This LCO is being added to remedy that situation. It will require that an evaluation be performed in the event that conductivity or pH are out-of-limit and cannot be brought back within limits within 72 hours. In addition, if the chlorides exceed specified limits in Table 3.6.B.2-1 and cannot be brought back into limits within 24 hours, an engineering evaluation must also be performed. The evaluation is made to determine if the structural integrity of the Reactor Coolant System remains acceptable for continued operation. This evaluation is to be performed and evaluated prior to leaving COLD SHUTDOWN. When the reactor is in a COLD SHUTDOWN condition, the reactor coolant is below 212°F with the system vented to the atmosphere. This will prevent the system from

being pressurized, which would increase the pressure of the piping and therefore, may increase the probability of crack growth or a leak in the system. This LCO will enhance the TS in that it imposes a new requirement for an engineering evaluation to ensure that the structural integrity of the Reactor Coolant System has not been jeopardized. This LCO also provides a shutdown statement if the engineering evaluation cannot ensure the structural integrity of the Reactor Coolant System.

3.6-3 Revise existing SR 4.6.B.1.a

Revise the existing SR as follows:
"~~A sample of reactor coolant shall be taken at least every 96 hours and analyzed for gross gamma activity of the filtrate from a 0.45 μ filter and gross iodine activity. I-131 and I-133 shall be determined weekly.~~ **The specific activity of the reactor coolant shall be demonstrated to be within limits by performance of the sampling and analysis program of Table 4.6.B.1-1.**"

This SR has been revised and incorporated into proposed Table 4.6.B.1-1. Placing this information into a tabular format presents it in a clear and concise manner. The sample frequency has changed from 96 hours to 72 hours. This proposed change is consistent with the guidance provided by the Standard TS.

3.6-3 Revise existing SR 4.6.B.1.b

Revise the existing SR as follows:
"~~Isotopic analysis via gamma spectrometry of a sample of reactor coolant shall be made at least once per month and will include I-131, I-132, I-133 and I-135.~~"

This SR has been revised and incorporated into proposed Table 4.6.B.1-1. Placing this information into a tabular format presents it in a clear and concise manner. This proposed change is consistent with the guidance provided in the Standard TS.

3.6-3 Revise existing SR 4.6.B.1.c

Revise the existing SR as follows:
"~~When the equilibrium iodine value is greater than or equal to 0.012 μCi/gm of dose equivalent I-131 a sample shall be taken prior to~~

~~startup and at 4-hour intervals during startup (minimum of 3 samples, maximum of 12) and analyzed for gross gamma activity of the filtrate.~~"

This SR has been revised and incorporated into the proposed Table 4.6.B.1-1. Placing this information into a tabular format presents it in a clear and concise manner. This proposed change is consistent with the guidance provided in the Standard TS.

3.6-3 Revise existing SR 4.6.B.1.d

Revise the existing SR as follows:
"~~When the equilibrium iodine value is greater than or equal to 0.012 μCi/gm of dose equivalent I-131 and the gaseous waste monitor located prior to holdup indicates an increase of greater than 50% in the steady state fission gas release after factoring out increases due to power changes a sample of reactor coolant shall be taken and a specific determination shall be made for I-131 dose equivalent for the iodine mixture.~~"

This SR has been revised and incorporated into the proposed Table 4.6.B.1-1. Placing this information into a tabular format presents it in a clear and concise manner. This proposed change is consistent with the guidance provided in the Standard TS.

3.6-3a Revise existing SR 4.6.B.1.e

Revise the existing SR as follows:
"~~When the equilibrium iodine value is greater than or equal to 0.012 μCi/gm of dose equivalent I-131 and following a significant power change** samples of reactor coolant shall be taken at 4 hour intervals (minimum of 3 samples, maximum of 12). If the significant power change occurs during normal working hours for laboratory personnel the samples shall be analyzed for gross iodine on the same working day. If the significant power change occurs during non-working hours for laboratory personnel the samples shall be taken by station personnel and analyzed for gross iodine during the next normal working day for laboratory personnel. **For the purpose of this section on sampling frequency, a significant power change is defined as a change exceeding 15% of rated power in less than 1 hour.~~"

The sample requirements in this SR have been revised and

incorporated into the proposed Table 4.6.B.1-1. Placing this information into a tabular format presents it in a clear and concise manner. In addition, the requirement pertaining to the sampling during off hours is being deleted. The DAEC lab personnel currently work shifts to support the operations of the plant. Samples can be obtained and analyzed as needed.

The footnote is being deleted. The revised SRs and the proposed Table provide adequate information as to when to obtain a sample.

3.6-3a Revise existing SR 4.6.B.1.f

Revise the existing SR as follows:
"~~When two successive samples required in c & e above indicate a decreasing trend below the limiting value of 1.2 μCi/gm dose equivalent I-131 the sampling shall be discontinued, (minimum of 3 samples).~~"

As stated above, the sampling requirements have been revised and incorporated into the Table 4.6.B.1-1. The requirements in this Table are consistent with the guidance provided in the Standard TS.

3.6-3a Revise existing SR 4.6.B.1.g

Revise the existing SR as follows:
"~~If, on the basis of gross activity measurements in a, c and e above, there is an indication that 1.2 μCi/gm, I-131 equivalent may be exceeded, a specific determination shall be made for I-131 dose equivalent for the iodine mixture.~~"

Again, the proposed Table 4.6.B.1-1 contains the surveillance requirements and frequencies for sampling the Reactor Coolant System.

3.6-3b Revise existing SR 4.6.B.1.h to new SR 4.6.B.1.b

Revise the existing SR as follows:
" Whenever the ~~I-131~~ ~~dose equivalent~~ **DOSE EQUIVALENT I-131** ~~as determined in (e) above~~ exceeds 0.6 μCi/~~gm~~ **ml**, ~~(50% of the equilibrium value)~~ notify the USNRC as specified **by** ~~in~~ 6.11.1.h."

An editorial change was made using all caps for "DOSE EQUIVALENT."

In addition, "**DOSE EQUIVALENT**" and "**I-131**" have been changed around.

The words, "as determined in (e) above" are being deleted. SR 4.6.B.1.e no longer exists; it has been replaced by information in Table 4.6.B.1-1.

The words "(50% of the equilibrium value)" are also being deleted. Again, the Table provides adequate guidance as to the requirements. This information is not needed.

The unit of measurement "gm" is being replaced with "**ml**." DAEC chemistry uses milliliters as a unit of measure. This change in units will not change the intent or any limits in the existing TS. In a conversion using water, 1 gram is equal to 1 ml. This conversion is also true when using reactor coolant water.

The word "**by**" replaces the word "in." This is an editorial change, made consistently throughout the section.

3.6-3b Revise the existing SR 4.6.B.2

Revise the existing SR as follows:
"~~A sample of~~ **The** reactor coolant shall be ~~analyzed~~ **determined to be within the specified chemistry limits by:**"

The changes are editorial by deleting the words "A sample of" and "analyzed" and adding the words "**The**" and "**determined to be within the specified chemistry limits by:**." These proposed changes do not change the intent. They do clarify the existing SR and incorporate the guidance provided in the Standard TS.

3.6-3b Revise the existing SR 4.6.B.2.a to Proposed 4.6.B.2.a and 4.6.B.2.b

Revise the existing SR as follows:
"~~At least every 4 hours during startup and at steaming rates below 100,000 pounds per hour for chloride ion content if the conductivity is above 0.5 μmho/cm or if it increases at a rate of 0.2 μmho/cm/hr or more. The minimum frequency will be once per day.~~"

Add SR 4.6.B.2.a

"**Measurement prior to pressurizing the reactor during each startup,**

if not performed within the previous 72 hours."

Add SR 4.6.B.2.b

"Obtain and analyze a sample of the reactor coolant at least once every 72 hours for chlorides and conductivity.*"

"* Not applicable with no fuel in the reactor vessel."

This SR has been revised and divided into the proposed SR 4.6.B.2.a and 4.6.B.2.b. The proposed SRs do not utilize the 100,000 pounds per hour. Instead the SRs utilize terms that are consistent with the ones generally seen in the TS. In addition, the proposed samples are taken prior to startup provided that they have not been taken in the previous 72 hours prior to startup and once every 72 hours when fuel is in the reactor vessel, provided the chlorides and conductivity are within specified limits. In the event that either is outside the limits, then samples will be taken and analyzed every eight hours as noted in proposed SR 4.6.B.2.c.

3.6-4 Revise existing SR 4.6.B.2.b

Revise the existing SR as follows:
"~~At least every 4 days when fuel is in the reactor vessel for conductivity and chloride ion content.~~"

This SR is also incorporated into SRs 4.6.B.2.a and 4.6.B.2.b. The existing SR requires a sample to be obtained and analyzed every 4 days whereas the proposed SRs require this to be performed once every 72 hours. The 72 hours is consistent with current industry practice and the guidance provided in the Standard TS.

Add new SR 4.6.B.2.c

Add the following SR:
"Obtain and analyze a sample of the reactor coolant for chlorides at least once every 8 hours whenever conductivity is greater than the limit specified in Table 3.6.B.2-1."

The existing TS do not include a requirement to sample and analyze reactor coolant for chlorides in the event that the conductivity exceeds the limits specified in Table 3.6.B.2-1. This proposed

change is an enhancement to the existing TS providing additional and clear guidance for sampling chlorides. This change is consistent with the guidance provided in the Standard TS.

Add new SR 4.6.B.2.d

Add the following SR:
"Obtain and analyze a sample of the reactor coolant for pH at least once every 8 hours whenever conductivity is greater than the limit specified in Table 3.6.B.2-1."

The existing TS do not include a requirement to sample and analyze reactor coolant for pH in the event that the conductivity exceeds the limits specified in Table 3.6.B.2-1. This proposed change is an enhancement to the existing TS providing additional and clear guidance for sampling pH. This change is consistent with the guidance provided in the Standard TS.

3.6-4 Revise existing SR 4.6.B.3.a to new SR 4.6.B.2.e

Revise the existing SR as follows:
"~~Conductivity is to be continuously monitored in three places: Reactor Water Cleanup System, between the hot well and the demineralizer beds, and at the outlet of the demineralizer beds.~~"

Replace with:
"With no continuous recording conductivity monitor OPERABLE, obtain an in-line conductivity measurement at least once per 4 hours when in RUN, STARTUP, or HOT SHUTDOWN MODES and 24 hours at all other times."

The proposed replacement is basically the same. This SR supports the addition of the proposed LCO 3.6.B.2.a.3. The locations of the three monitors are identified in the existing SR. These locations have been relocated to the Bases Section. The requirement that the in-line monitor be installed has also been added to the SR. The proposed SR provides more guidance as to actions to take in the event that continuous monitoring is not available. This change is consistent with the guidance provided in the Standard TS.

3.6-4 Revise existing SR 4.6.B.3.b

Revise the existing SR as follows:
"~~In the event that one of the three monitors becomes inoperable, conductivity is to be measured and recorded with a temporary instream monitor.~~"

This SR has been incorporated for the most part in the proposed SR 4.6.B.2.e.

Add SR 4.6.B.2.f
"Perform a CHANNEL CHECK of the continuous conductivity monitor at least once per 7 days."

This SR is being added to ensure that the conductivity monitor is OPERABLE. The existing SR does not include this surveillance.

DAEC TS Section 3/4.6.C, Coolant Leakage
The following are proposed changes to the DAEC Coolant Leakage Section of the TS.

3.6-5 Revise existing LCO 3.6.C.1

The existing LCO is being revised as follows:
"~~Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F~~ **When in RUN, STARTUP, or HOT SHUTDOWN MODE, the** ~~r~~ Reactor ~~c~~ Coolant **System** leakage into the ~~primary containment~~ **drywell** shall be limited to:"

The phrase, "Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F" is being replaced with, **"When in RUN, STARTUP, or HOT SHUTDOWN MODE, the... ."** One of the main purposes of this Section 3.6 rewrite is to provide clarity and consistency throughout the entire section. One of the consistency changes is to include references to the applicable MODES as defined in the DAEC TS. The addition of these MODES do not change the intent or operating conditions the plant is currently using. This proposed change makes the LCO clearer and consistent with the guidance provided by the Standard TS.

The following proposed changes are editorial:

The words "Reactor Coolant System" are being capitalized. This is being done for consistency throughout the DAEC TS.

The word **"System"** is being added to provide clarity and consistency throughout the DAEC TS Section 3.6.

The words "primary containment" are being replaced with **"drywell."** The DAEC terminology is drywell as noted in the UFSAR. Any leakage from the Reactor Coolant System would be contained in either the equipment or floor drain sumps which are located in the drywell. Using the word **"drywell"** makes the TS technically correct as applied to DAEC.

Reliable means are provided to detect leakage from the nuclear system barrier inside the drywell. Nuclear system leakage rate limits are established 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. This capability includes the following:

1. Identifying the reactor primary system leakage sources.

2. Efficiently isolating and controlling the sources.

3. Effectively removing the residual leakage water.

4. Conveniently replacing the leakage liquid and/or restoring the source system function.

These functions are accomplished under normal operation or postaccident conditions in a manner such 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 are not impaired or negated.

The leakage considered here is limited to water or steam released from the nuclear system process barrier inside the primary containment (drywell). 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 release 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.

IDENTIFIED LEAKAGE into the equipment drain sump is composed of normal seal and valve packing leakage and does not represent a safety consideration so long as the leakage is small compared to the available reactor coolant makeup capability.

Most valves and pumps in the nuclear system inside the drywell are equipped with double seals; leakage through the primary seal is piped to the equipment drain sump.

Leakage from the reactor vessel head flange gasket is piped to a collection chamber and then to the equipment drain sump. The chamber filling time is periodically timed during plant operation,

and the flange gasket leakage rate is calculated.

Leakage from the main steam relief and safety valves is identified by downstream temperature sensors that read out in the main control room. Relief valve discharge is directed to the suppression pool.

UNIDENTIFIED LEAKAGE consists of all leakage from the reactor primary system that is not IDENTIFIED LEAKAGE. UNIDENTIFIED LEAKAGE is collected in the drywell floor drain sump. Vapor that is condensed by the drywell ventilation system will drain to this sump.

Six different methods are used to detect leakage in the primary containment:

1. Equipment drain sump flow.

2. Floor drain sump flow.

3. Drywell ventilation system cooling water temperature.

4. Drywell pressure.

5. Drywell temperature.

6. Drywell atmosphere radioactivity.

The different drywell parameters listed above 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. When IDENTIFIED LEAKAGE is combined with UNIDENTIFIED LEAKAGE, the TOTAL LEAKAGE limit becomes 25 gpm.

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 each.

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 Control Rod Drive (CRD) and Reactor Core Isolation Cooling (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.

3.6-5 Revise existing LCO 3.6.C.1.a

The existing LCO is being revised as follows:
"≤ 5 gpm ~~unidentified leakage~~ **UNIDENTIFIED LEAKAGE.**"

The existing TS states a 5 gpm unidentified leakage limit. The "≤" sign has been added to provide clarification noting that the limit is a range up to and including 5 gpm. The existing TS inferred this, but, by adding the "≤" sign, any possible confusion is eliminated.

UNIDENTIFIED LEAKAGE is a defined term as submitted by this proposed amendment request. Defined terms in the DAEC TS appear in all caps. Again, this is an editorial change to provide consistency throughout the DAEC TS.

3.6-5 Revise existing LCO 3.6.C.1.b

The existing LCO is being revised as follows:
"≤ 2 gpm increase in ~~unidentified leakage~~ **UNIDENTIFIED LEAKAGE** within a 24 hour period."

The existing TS states a specific limit of 2 gpm increase in UNIDENTIFIED LEAKAGE within a 24 hour period. The "≤" sign has been added to provide clarification noting that the limit is a range up to and including 2 gpm. The existing TS inferred this, but by adding the "≤" sign eliminates any possible confusion. The requirement that an increase in UNIDENTIFIED LEAKAGE shall not exceed 2 gpm in a 24 hour period is based on maintaining the ability to detect small leaks in a reasonably short time so that corrective

action can be initiated. However, during reactor startup and ascension to normal operating pressure, leakage should be closely monitored until normal operating pressure is achieved and a "baseline" leakage rate can be established to which the leakage increase can be compared. Once established, the "baseline" will be used; therefore, if UNIDENTIFIED LEAKAGE exceeds 2 gpm over the "baseline," the appropriate actions shall be taken.

UNIDENTIFIED LEAKAGE is a defined term added by this amendment request. Defined terms in the DAEC TS appear in all caps.

In addition the word, "hour" is spelled out instead of using the abbreviation, "hr." These are editorial changes to provide consistency throughout the TS.

3.6-5 Revise existing LCO 3.6.C.1.c

The existing LCO is being revised as follows:
"≤ 25 gpm ~~total leakage~~ **TOTAL LEAKAGE.**"

The existing TS states a 25 gpm total leakage limit. The "≤" sign has been added to provide clarification noting that the limit is a range up to and including 25 gpm. The existing TS inferred this, but by adding the "≤" sign, any possible confusion is eliminated.

TOTAL LEAKAGE is a defined term in the DAEC TS. Defined terms are in all caps in the DAEC TS. This is considered an editorial change.

3.6-5 Revise existing LCO 3.6.C.3 to new 3.6.C.2

The existing LCO is being revised as follows:
"~~If~~ **With** the conditions in ~~1 or 2~~ **Specifications 3.6.C.1.a, b, or c above** ~~cannot be~~ **not** met, ~~an orderly shutdown shall be initiated and the reactor shall be~~ **reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and** in ~~a Cold Shutdown~~ **COLD SHUTDOWN** ~~Condition~~ within **the following** 24 hours."

The word "if" is being replaced by the word "**With.**" This is an editorial change being made to provide consistency throughout the TS section.

The TS LCO numbers "**Specifications 3.6.C.1.a, b, or c above**" are

replacing the numbers "1 or 2" to specifically identify the leakage limits specified in LCO 3.6.C.1. This proposed change is editorial, enhances the TS LCO and eliminates any potential confusion as to what LCO is applicable.

The words "cannot be" are being replaced with the word "**not.**" This is an editorial change to make the sentence more grammatically correct.

The existing TS LCO states, "an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours." This is being revised to state, "**reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and** in **COLD SHUTDOWN** within **the following** 24 hours." The existing TS do not contain any time interval to bring the leakage back to within limits before initiating shutdown action. This would require the operators to prematurely initiate a reactor shutdown.

The proposed change revises a requirement to shutdown when the LCO limit on Reactor Coolant System leakage is exceeded by allowing a 4 hour period to bring the leakage back within limits. If the leakage cannot be reduced within the required time, the standard shutdown requirement to be in HOT SHUTDOWN and eventually COLD SHUTDOWN is initiated. The 4 hour period is a justified time frame accepted by the NRC. The 4 hours is adequate time to allow the reduction in leakage to be brought into compliance with the limits as specified in the TS. The probability of an accident exceeding the safety analysis during this 4 hour period is minimal and the change therefore, does not increase the probability or 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 proposed change introduces the action of being in HOT SHUTDOWN within 12 hours if the leakage limits cannot be restored to within limits within 4 hours. The requirement to be in COLD SHUTDOWN within 24 hours was maintained from the existing TS. This proposed action is consistent within the industry as provided by the guidance in the Standard TS.

3.6-5 <u>Revise existing LCO 3.6.C.2 to new LCO 3.6.C.3</u>

The existing LCO is being revised as follows:
"When in RUN, STARTUP, or HOT SHUTDOWN MODE, ~~I~~ **t**he ~~s~~ Sump ~~s~~ System shall be ~~operable~~ **OPERABLE as defined in Table 3.2-E.** ~~any time irradiated fuel is in the vessel and reactor coolant temperature is above 212°F. From and after the date that the sump system is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 24 hours unless the system is made operable sooner, provided the air sampling system is operable.~~"

The phrase, "any time irradiated fuel is in the vessel and reactor coolant temperature is above 212°F" is being replaced with, **"When in RUN, STARTUP, or HOT SHUTDOWN MODE,... ."** One of the main purposes of this Section 3.6 rewrite is to provide clarity and consistency throughout the entire section. One of the consistency changes is to replace language describing shutdown requirements with applicable MODES as defined in the DAEC TS. The reference to MODES does not change the intent or operating conditions the plant is currently using. This proposed change makes the LCO clearer and is consistent with the guidance provided by the Standard TS.

The **"Sump System"** is being editorially changed to initial caps. This is a system as stated in the DAEC TS. This change is being made for consistency throughout the DAEC TS.

The word, **"OPERABLE"** is in all caps since it is a defined term in the DAEC TS. This change is being made for consistency throughout the DAEC TS Section 3.6.

The words **"as defined in Table 3.2-E"** are being added to specifically define the Sump System OPERABILITY requirements. The rest of the existing LCO is being divided into proposed LCOs 3.6.C.4 and 3.6.C.5 below. The LCO provides specific and additional guidance to the operators as to what system OPERABILITY consists of and what required actions are to be taken when the systems are not OPERABLE.

<u>Add LCO 3.6.C.4</u>

Add the following LCO:
"With the Sump System inoperable, immediately verify the Air Sampling System is OPERABLE and restore the Sump System to OPERABLE status within the next 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following

24 hours."

This LCO is basically a rewrite of the existing LCO 3.6.C.3. When the Sump System is inoperable, the Air Sampling System is immediately to be verified OPERABLE. This proposed specification provides specific actions and shutdown requirements to be taken in the event the Sump System is inoperable. This LCO allows plant operation with an inoperable Sump System for 24 hours provided that the Air Sampling System is OPERABLE. If the Sump System is not restored to OPERABLE status within 24 hours, then a reactor shutdown is initiated so that the plant is in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. This change provides consistency with the guidance provided in the Standard TS.

Add LCO 3.6.C.5

Add the following LCO:
"With both the Sump System and the Air Sampling System inoperable, restore one of the systems to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours."

This LCO is being added to cover the condition where both the Sump System and the Air Sampling System are inoperable. The existing TS have no LCO or action requirements for this condition. When both systems are inoperable, there is a 4 hour time limit to restore at least one system to OPERABLE status. If at the end of the 4 hours neither system is OPERABLE, then a reactor shutdown action is initiated. The shutdown requirement is consistent with the guidance provided in the Standard TS shutdown requirements. The addition of this revised LCO provides clarity and consistency with the rest of the DAEC TS Section 3.6.

3.6-5 Revise existing SR 4.6.C.1

The following SR is being revised as follows:
"Reactor ~~c~~ **C**oolant ~~s~~ **S**ystem leakage shall be checked by the ~~s~~ **S**ump ~~s~~ **S**ystem and recorded at least once every 8 hours."

This proposed change capitalizes "Reactor Coolant System" and "Sump System." This is an editorial change which also is consistent with the rest of the DAEC TS in this section.

Add new SR 4.6.C.2

Add the following SR:
"Verify Sump System OPERABILITY as specified in Table 4.2-E."

This SR is being added to ensure Sump System OPERABILITY. The existing SRs do not contain any such requirements. The addition of this SR eliminates that deficiency and enhances the overall TS.

3.6-5 Revise existing SR 4.6.C.2 to new SR 4.6.C.3

The existing SR is being revised as follows:
"Verify Air Sampling System OPERABILITY as specified in Table 4.2-E. The ~~a~~ **A**ir ~~s~~ **S**ampling ~~s~~ **S**ystem shall be checked and recorded at least once every 8 hours."

The existing TS does not require verification of the Air Sampling System OPERABILITY. The proposed revision adds the sentence, **"Verify Air Sampling System OPERABILITY as specified in Table 4.2-E."** Existing Table 4.2-E contains the SR that ensures the Air Sampling System is OPERABLE. Adding this SR eliminates the noted deficiency.

In addition, the words **"Air Sampling System"** are being capitalized. The names of systems are being capitalized to provide consistency throughout this section. This is an editorial change and does not change the intent of the SR.

DAEC TS Section 3/4.6.D, Safety and Relief Valves

The following are proposed changes to the DAEC Safety and Relief Valves Section of the TS.

3.6-5 Revise existing LCO 3.6.D.1

The existing LCO is being revised as follows:
"~~During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F~~ **When in RUN, STARTUP, or HOT SHUTDOWN MODE**, both safety valves and the safety modes of all relief valves* shall be ~~operable~~ **OPERABLE**, except as specified in **Specification** 3.6.D.2."

"*SRVs which perform an ADS function must also satisfy the OPERABILITY requirements of Specification 3.5.F, Core and Containment Cooling Systems."

The phrase, "During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F ..." is being replaced with, "**When in RUN, STARTUP, or HOT SHUTDOWN MODE ... .**" This change is basically editorial. One of the main objectives of rewriting this section is to provide consistency and clarity throughout. Using the proper defined MODES of operation instead of the language currently in the DAEC TS fulfills this objective. Using the proper MODES does not change the intent or any operating conditions.

The footnote "***SRVs which perform an ADS function must also satisfy the OPERABILITY requirements of Specification 3.5.F, Core and Containment Cooling Systems.**" is being added to this LCO. It provides additional guidance, informing Operations that there is another TS to be met in the event a relief valve is inoperable.

The word "**OPERABLE**" is to be in all caps since it is a defined term in the DAEC TS.

The word "**Specification**" is being added as an editorial change to provide consistency throughout DAEC TS Section 3.6.

3.6-6 Revise existing LCO 3.6.D.2.a

The existing LCO is being revised as follows:

"~~From and after the date that~~ **With** the safety valve function of one relief valve ~~is made or found to be~~ inoperable, ~~continued reactor operation is permissible only during the succeeding~~ **restore the inoperable safety valve function to OPERABLE status within** thirty days. ~~unless such valve function is sooner made OPERABLE.~~"

The phrase "From and after the date that" is being replaced by the word "**With.**" This editorial change provides clarity and is consistent with the other LCOs in this section and with the guidance provided in the Standard TS.

Deleting the words "is made or found to be" will not change the intent of the TS. This proposed change deletes superfluous wording. It does not matter if the valve is found or made to be inoperable. The fact is that the valve will not perform its intended function despite how it became inoperable.

The phrase "continued reactor operation is permissible only during the succeeding" is being replaced by "**restore the inoperable safety valve function to OPERABLE status within.**" The intent has not been changed. The proposed wording is consistent with LCO 3.6.D.2.b and with the guidance provided by the Standard TS.

The words "unless such valve function is sooner made OPERABLE" are being deleted. In this LCO, the basic intent is that if the valve function is not restored within the thirty day period a reactor shutdown is initiated. If the valve function is restored to operation within thirty days, then continued reactor operation is permissible. It has always been the intent of the TS that if a piece of equipment is restored to OPERABLE status within the LCO time frame, then the LCO may be exited and no other actions are required.

The existing LCO did not contain any specific action statement in the event that the safety valve function of one relief valve became inoperable. Proposed LCO 3.6.D.3 contains such shutdown requirements.

3.6-6 <u>Revise existing LCO 3.6.D.2.b</u>

The existing TS is being revised as follows:
"~~From and after the date that~~ **With** the safety valve function of two relief valves ~~is made or found to be~~ inoperable, ~~continued reactor operation is permissible only during the succeeding~~ **restore the**

inoperable safety valve function to OPERABLE status within seven days ~~unless such valve function is sooner made OPERABLE~~."

The phrase "From and after the date that" is being replaced by the word **"With."** This editorial change provides clarity and is consistent with the other LCOs in this section and with the guidance provided by the Standard TS.

Deleting the words "is made or found to be" will not change the intent of the TS. This proposed change deletes superfluous wording. It does not matter if the valve is found or made to be inoperable. The fact is that the valve will not perform its intended function despite how it became inoperable.

The phrase "continued reactor operation is permissible only during the succeeding" is being replaced by **"restore the inoperable safety valve function to OPERABLE status within."** The intent is the same and the proposed wording is consistent with LCO 3.6.D.2.a and the guidance provided by the Standard TS.

The words "unless such valve function is sooner made OPERABLE" are being deleted. In this LCO, the basic intent is that, if the valve function is not restored within the seven day period, a reactor shutdown is initiated. If the valve function is restored to operation within the 7 days, then continued reactor operation is acceptable. It has always been the intent of the TS that, if a piece of equipment is restored to OPERABLE status within the LCO time frame, then the LCO may be exited and no other actions are required. The shutdown requirement is located in proposed LCO 3.6.D.3, which is consistent with proposed LCO 3.6.D.2.a above.

3.6-6 <u>Revise existing LCO 3.6.D.3</u>

The existing TS is being revised as follows:
"If Specification 3.6.D.1 **or 3.6.D.2** is not met, ~~an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to atmospheric within 24 hours~~ **be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.**"

The reference to proposed LCO 3.6.D.2 is being added since LCO 3.6.D.3 provides the shutdown requirements in the event the referenced LCOs cannot be met. This is the standard shutdown requirement which is being used throughout revised Section 3.6.

The phrase, "an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to atmospheric within 24 hours" is being replaced with, **"be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours."** This change is editorial in nature. The intent of the initial shutdown requirement is basically the same except for the added step of being in at least HOT SHUTDOWN within the next 12 hours. This shutdown requirement is consistent with the other shutdown requirements being added to this section in addition to being consistent with the guidance provided by the Standard TS.

3.6-5 Revise existing SR 4.6.D.1

The existing SR is being revised as follows:
"~~At least~~ **Once per OPERATING CYCLE, at least** one safety valve and 3 relief valves shall be ~~checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles.~~ **removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested. The safety and relief valves shall be rotated, at least once per 40 months, such that both safety and 6 relief valves are removed, set pressure tested and reinstalled or replaced with spares. Any spare that is installed must have been set pressure tested within the previous 40 months.**

The setpoint of the safety valves shall be as specified in Specification 2.2."

This proposed revision expands the SR to come in closer alignment with the guidance provided in the Standard TS. The words **"Once per OPERATING CYCLE"** are being moved to the beginning of the sentence and the words "at least" will proceed. This is an editorial change in that the existing SR already specifies this frequency. OPERATING CYCLE is a defined term in the DAEC TS and therefore appears in all caps. This is consistent with other SRs in this section as well as with the guidance provided by the Standard TS.

The SR will preserve the requirement to remove and test one safety valve and 3 relief valves each OPERATING CYCLE. DAECs design includes 2 safety and 6 relief valves. These requirements are consistent with the rest of the industry. Standard TS require that 1/2 of the safety and relief valves be removed and tested each OPERATING CYCLE.

The existing SR continues to state that, "... checked or replaced with bench checked valves ... . All valves will be tested every two cycles." These requirements have been revised to state, **"removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested. The safety and relief valves shall be rotated, at least once per 40 months, such that both safety and 6 relief valves are removed, set pressuro tested and reinstalled or replaced with spares."** This new requirement is consistent with the existing TS except the periodicity is now 40 months vs. 36 months. The additional 4 months has been previously accepted by the NRC in the Standard TS. Testing or replacement of these valves must be done when the plant is shutdown (REFUELING MODE). To assist utilities, the NRC approved the 40 months thereby ensuring that plants can meet the SR limits without requiring any extensions in the event that the plant does not enter its refueling outage within the typical 18 month window. In addition, the proposed SR places established requirements for those safety and relief valves that are in storage: **"Any spare that is installed must have been set pressure tested within the previous 40 months."** This requirement ensures that any spares installed meet the SR when installed.

The proposed Bases Section has been expanded to state that these valves will be set pressure tested and stored in accordance with manufacturer's recommendations. There may be conditions where DAEC may not be notified by the manufacturer of new storage requirements or DAEC may take exception with the requirements. In these isolated cases, DAEC and the manufacturer will come to resolution on an acceptable position.

3.6-6 <u>Revise existing SR 4.6.D.2</u>

The existing SR is being revised as follows:
"At least one of the relief valves shall be disassembled and inspected once per ~~operating cycle~~ **OPERATING CYCLE**."

This proposed change is editorial in that the words **"OPERATING CYCLE"** are all in caps because it is a defined term in the DAEC TS. This is being made throughout the section for consistency. It does not change the intent or requirements of the existing SR. This SR is being maintained due to a commitment made in the DAEC ISI Section XI Testing Program.

3.6-6 Revise existing SR 4.6.D.3

The existing SR is being revised as follows:
"With the reactor pressure ≥100 psig and turbin**e** ~~turbing~~ bypass flow to the main condenser, each relief valve shall be manually opened and verified open by turbine bypass valve position decrease**,** ~~and~~ pressure switches and thermocouple readings downstream of the relief valve to indicate steam flow from the valve once per ~~operating cycle*~~ **OPERATING CYCLE.**

~~* Intent Change Only (definition of operating cycle).~~"

The proposed changes in this SR are editorial. The word, "turbine" is incorrectly spelled in the existing TS. A comma is being added after the word "decrease." This is to make the sentence grammatically correct since there are more than two items in the listing.

The word "and" is being deleted, again to make the sentence grammatically correct. The sentence contains three items in a series with the first two items separated by a comma and the last item joined by the word "and".

"**OPERATING CYCLE**" appears in all caps since it is a defined term. This is an editorial change which is being made throughout Section 3.6.

The footnote," *Intent Change Only (definition of operating cycle)" is being deleted. This footnote is no longer needed. The wording of the SR is clear without the footnote. This does not change the intent or requirements of the existing SR.

3.6-6 Revise existing SR 4.6.D.4

The existing SR is being revised as follows:
"The relief valve setpoints for the Low-Low Set function shall be as specified in Section 2.2.1.c. Instrumentation and system logic shall be functionally tested, calibrated, and checked as ~~required~~ **specified** in Table 4.2-B."

This proposed change is editorial. The word "**specified**" replaces the word "required." The word "**specified**" provides clarification and makes the sentence consistent with other TS in Section 3.6. This does not change the intent or requirements of the existing SR and is consistent with the guidance provided by the Standard TS.

DAEC TS Section 3/4.6.E, Jet Pumps
The following are proposed changes to the DAEC Jet Pumps Section of the TS.

3.6-6a Revise existing LCO 3.6.E.I

The existing LCO is being revised as follows:
"Whenever the reactor is in the RUN **or STARTUP MODE** ~~mode~~, all jet pumps shall be OPERABLE. ~~If the requirements of 4.6.E.1.a or .b are not met, perform the surveillance requirements of 4.6.E.2 within 24 hours. If one or more jet pumps do not meet the requirements of 4.6.E.2 and~~"

Reference to an additional MODE **"STARTUP"** is being added. **"STARTUP"** is a defined term and appears in all caps. In addition, the word **"or"** was added, instructing the operator that when in either the RUN or STARTUP MODES that the jet pumps are to be OPERABLE. The jet pumps are required to be OPERABLE in RUN and STARTUP MODES since there is a large amount of energy in the reactor core and since the limiting design basis accidents are assumed to occur in either of these MODES. This is also consistent with the requirements for operation of the Recirculation System as specified in LCO 3.3.F.1.

The word **"MODE"** is capitalized. All defined terms are to appear in caps when used in the DAEC TS. This is an editorial change providing consistency throughout the DAEC TS Section 3.6.

The second sentence of the LCO is being relocated in its entirety to the proposed SR 4.6.E.1.c.

3.6-6a Revise existing LCO 3.6.E.1 to new 3.6.E.1.a

The existing LCO is being revised as follows:
"If one or more jet pumps do not meet the **Surveillance** ~~r~~ **R**equirements of 4.6.E.2 ~~and~~ **with:**"

The following are editorial changes:

The referenced number 4.6.E.2 is a Surveillance Requirement. To ensure clarity and consistency throughout the DAEC TS Section 3.6, the word **"Surveillance"** is being added in initial caps. In addition, "**R**equirements" is to appear in initial caps. This is an editorial change.

The word "and" is being replaced with the word "**with**." The proposed format is to divide the existing LCO into proposed LCOs 3.6.E.1.a.1 and 3.6.E.1.a.2 as part of LCO 3.6.E.1.a.

3.6-6a Revise existing LCO 3.6.E.1.a to new LCO 3.6.E.1.a.1

The existing TS is being revised as follows:
"the recirculation pump speed ~~is~~ less than 60% of rated, continue to monitor the jet pump(s) performance per **Surveillance Requirement** 4.6.E.2 daily until the evaluation can be performed at pump speed greater than 60% **of rated**."

The following changes are editorial:

The word "is" is being deleted as superfluous. This deletion is to provide clarity and does not change the intent of the LCO.

The words "**Surveillance Requirement**" appear in initial caps and are being added preceding the number 4.6.E.2. This change is purely editorial to provide additional clarity.

The words "**of rated**" are being added to provide clarity and are consistent with proposed LCO 3.6.E.1.a.2 and SRs 4.6.E.4.

3.6-6b Delete blank page

This page is being deleted. It does not contain any information and was used as a filler page for this section.

3.6-7 Revise existing LCO 3.6.E.1.b to new LCO 3.6.E.1.a.2

The existing TS is being revised as follows:
"the recirculation pump speed ~~is~~ greater than or equal to 60% of rated, evaluate the reason for the deviation ~~and~~. If the evaluation verifies the jet pump(s) to be ~~INOPERABLE~~ **inoperable, be in at least HOT SHUTDOWN within the next 12 hours** ~~the reactor shall be placed in COLD SHUTDOWN within 24 hours~~."

The following proposed changes are editorial:

The word "is" is being deleted as a superfluous word.

The LCO has been split into two sentences which deleted the word "and" and capitalized the word "If."

The word "inoperable" is not a defined term in the DAEC TS and is not to be capitalized.

The existing action for this LCO includes a shutdown statement that if the conditions of the LCO cannot be met, then the plant is to be placed in COLD SHUTDOWN within 24 hours. This shutdown requirement is being replaced with the following shutdown requirement statement, "**... be in at least HOT SHUTDOWN within the next 12 hours.**" The new shutdown action still allows the reactor to be shutdown in an orderly manner. When a piece of equipment is inoperable and not returned to service within the LCO/Action time period, the reactor is to be placed in a MODE of operation in which the piece of equipment is not required to be OPERABLE. In this case, placing the reactor in HOT SHUTDOWN fulfills this requirement. Placing the reactor in COLD SHUTDOWN is not needed and requires the operators to cycle the plant to colder temperatures unnecessarily. This action and time limit for achieving HOT SHUTDOWN are consistent with current industry practice and with the guidance provided by the Standard TS. .

3.6-6a <u>Revise existing SR 4.6.E.1</u>

The existing SR is being revised as follows:
"Jet pump ~~operability~~ **OPERABILITY** shall be verified daily, following startup of a recirc**ulation** pump and after any unexplained changes in either ... ."

The following proposed changes are editorial:

The word "**OPERABILITY**" is a defined term in the DAEC TS and is to be capitalized.

The abbreviation for recirculation has been used. For consistency and to avoid any misinterpretations, it is to be spelled out as noted above.

3.6-6a <u>Revise SR 4.6.E.1.a</u>

The existing SR is being revised as follows:
"~~I~~ **t**he recirculation pump flow to ... ."

The revision to this SR is strictly editorial. The "T" in word "**t**he" is being made lower case since it is a continuation of the sentence from SR 4.6.E.1.

3.6-7 Revise SR 4.6.E.1.b

The existing SR is being revised as follows:
"~~T~~ **t**he jet pump loop flow ... ."

The revision to this SR is strictly editorial. The "T" in word "**t**he" is being made lower case since it is a continuation of the sentence from SR 4.6.E.1.

Add SR 4.6.E.1.c

Add the following SR:
"if the Surveillance Requirements of 4.6.E.1.a or 4.6.E.1.b are not met, perform the Surveillance Requirements of 4.6.E.2 within 24 hours."

This SR has been relocated from LCO 3.6.E.1. This is a surveillance requirement, not an LCO, and therefore should be relocated as proposed.

In addition, the following editorial changes have been made:

The word "**Surveillance**" has been added preceding the word "Requirements." This is to provide clarification and consistency.

The entire SR number "**4.6.E.1**.b" has been added again for clarification reasons only.

The words "**Surveillance Requirements**" appear in initial caps to be consistent throughout DAEC TS Section 3.6.

3.6-7 Existing SR 4.6.E.2

No changes were made to SR 4.6.E.2.

3.6-7 Revise existing SR 4.6.E.3

The existing SR is being revised as follows:
"The Surveillance Requirements of 4.6.E.1 and 4.6.E.2 do not apply to the ... ."

The following editorial changes are being made for clarity and provide consistency throughout the entire TS:

The words "Surveillance Requirements" appear in initial caps to be consistent throughout this section.

The entire SR number "4.6.E.2" is being added for clarity.

3.6-7 Existing SR 4.6.E.4

No changes were made to SR 4.6.E.4.

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

The following are proposed changes to the DAEC Jet Pump Flow Mismatch Section of the TS.

3.6-7 Revise existing LCO 3.6.F.1

The existing LCO is being divided into two separate itemized LCOs 3.6.F.1 and 3.6.F.2. This will provide additional clarity even though the LCO requirements have not changed. The first LCO incorporates the condition where core power is greater than or equal to 80% RATED POWER and the second LCO covers the condition where the core power is less than 80% RATED POWER. The existing TS covers both conditions in a single LCO statement which makes it confusing.

3.6-7 Revise existing LCO 3.6.F.1

The existing LCO 3.6.F.1 is being revised as follows:
"~~When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 122% of the speed of the slower pump when core power is 80% or more of rated power or 135% of the speed of the slower pump when core power is below 80% of rated power.~~ **With core power greater than or equal to 80% RATED POWER with both recirculation pumps at steady state operation, the speed of the faster pump may not exceed 122% of the speed of the slower pump.**"

As stated above, this LCO is basically the first part of the existing LCO 3.6.F.1. The following editorial changes have been made:

The word "When" has been replaced by "**With**".

The words "are" and "in" are being deleted.

These changes are minor editorial changes made to this LCO since it now stands alone. These minor editorial changes do not change the intent of the existing requirement.

Add new LCO 3.6.F.2

The existing LCO 3.6.F.1 is being revised as follows:
"**With core power less than 80% RATED POWER with both recirculation**

pumps at steady state operation, the speed of the faster pump may not exceed 135% of the speed of the slower pump."

Again, as stated above, this LCO is basically the second part of the existing LCO 3.6.F.1. There have only been some minor editorial changes made since it now stands alone. These minor editorial changes do not change the intent of the existing requirement.

Add new LCO 3.6.F 3

Add the following LCO:
"With the recirculation pump speeds different by more than the specified limits:"

This LCO is being added for consistency with LCO 3.6.F.3.a. Reference discussion for proposed LCO 3.6.F.3.a below.

Add new LCO 3.6.F.3.a

Add the following LCO:
"restore the recirculation pump speeds to within the specified limit within 2 hours, or"

This LCO is being added because the existing LCO requires that, if LCO 3.6.F.1 cannot be met, one recirculation pump shall be tripped and LCO 3.3.F.4 for single loop operation shall be complied with. The existing TS does not allow any time for the operators to try to restore the recirculation pump speed to within its limits before taking a specific action. Placing the plant into a single loop condition as soon as practical once outside the limits unnecessarily cycles the plant and is a less conservative action than to try to restore the pump, thus fixing the problem. In the event that the pump cannot be restored to within limits, the pump suspected to be out-of-limits is tripped and single loop operation is entered. The two hour completion time is based on the low probability of an accident occurring during this time period, a reasonable time to complete the action, and frequent core monitoring by operators allowing any abrupt changes in core flow conditions to be quickly detected.

3.6-7 Revise existing LCO 3.6.F.2 to new LCO 3.6.F.3.b

The existing LCO is being revised as follows:
"~~If Specification 3.6.F.1 cannot be met,~~ one recirculation pump shall be tripped. See Specification 3.3.F.4 for SLO requirements."

The revision will delete the words "If Specification 3.6.F.1 cannot be met." The reorganization makes the words unnecessary. Proposed LCO 3.6.F.3 incorporates the intent of the deleted words.

3.6-7 Revise existing SR 4.6.F.1

The existing SR is being revised as follows:
"Recirculation pump speed~~s~~ **mismatch** shall be ~~checked and logged~~ **verified** at least once per day."

The word "speed" is being made singular. This is an editorial change.

The word "**mismatch**" is being added to provide clarification and specifically state that the mismatch between the pumps is to be verified. This is editorial and consistent with the guidance provided by the Standard TS.

The words "checked and logged" are being replaced with the word "**verified**." Using "verified" does not change the intent of the existing SR. This change is being made to provide consistency within the DAEC TS Section 3.6 and is consistent with the guidance provided by the Standard TS.

3.6-7 Revise existing 4.6.F.2

The existing SR is being revised as follows:
"See ~~Specification~~ **Surveillance Requirement** ~~3~~**4**.3.F.4 for SLO requirements."

The word "Specification" is being replaced by the words "**Surveillance Requirement**." In addition, the SR references SR 4.3.F.4 for SLO which is consistent with the reference in the proposed SR. These are editorial changes which are being made for consistency throughout DAEC TS Section 3.6.

DAEC TS Section 3/4.6.G, Structural Integrity

The following are proposed changes to the DAEC Structural Integrity Section of the TS.

3.6-8 Revise existing LCO 3.6.G to LCO 3.6.G.1

The existing LCO is being revised as follows:
"At all times, ~~I~~the structural integrity of the ~~pressure boundaries~~ **ASME Section XI Code Class 1, 2, and 3 components** shall be maintained ~~at the level required by the original acceptance standard throughout the life of the plant~~ **in accordance with Surveillance Requirement 4.6.G.1."**

"At all times," is being added to make clear what existing TS imply; these requirements apply under all plant conditions. This addition provides clarity to the LCO and is consistent with the guidance provided by Standard TS.

The words "pressure boundaries" are being replaced with the wording **"ASME Section XI Code Class 1, 2, and 3 components."** This specifically identifies which components fall under this LCO. The existing TS do not provide a clear understanding of what is required and may leave the inclusion of some components up to interpretation.

The last part of the existing LCO "... at the level required by the original acceptance standard throughout the life of the plant" is being replaced with **"in accordance with Surveillance Requirement 4.6.G.1."** This change is for clarification. SR 4.6.G.1 references ASME Section XI which provides the requirements for maintaining the plant at an acceptance level throughout its life. Since SR 4.6.G.1 provides this information, repeating it in the LCO is not necessary. The proposed wording is also consistent with the guidance provided by the Standard TS.

Add new LCO 3.6.G.2

Add the following LCO:
"With the structural integrity of any ASME Section XI Code Class 1 or Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 212°F."

The existing TS does not specify the action to be taken when a Class 1 or Class 2 component does not comply with LCO 3.6.G.1. Adding this LCO action provides specific guidance when Class 1 or Class 2 component(s) do not conform to the ASME Section XI requirements. The main purpose of this TS is to ensure that structural integrity is maintained throughout the life of the plant. One major area of concern is under high pressure and low temperature conditions. Figure 3.6-1 provides NDT curves for operation of the plant. These curves are to be followed during operation by ensuring that the plant operates in the acceptable region of the curve. When in the REFUELING MODE, the curve sets a minimum temperature of about 75°F when the reactor is at atmospheric pressure. Allowing the plant temperature to increase but maintaining less than 212°F is acceptable in this condition. With the plant temperature less than 212°F and the pressure at atmospheric, there will not be a challenge to the structural integrity of any system. On the other hand, if the pressure increased, the potential for a leak might increase. This addition clarifies the existing TS and is consistent with the guidance provided by the Standard TS.

Add new LCO 3.6.G.3

Add the following LCO;
"With the structural integrity of any ASME Section XI Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service."

The existing TS do not specify action when a Class 3 component does not comply with LCO 3.6.G.1. Adding this LCO action provides that specific guidance if Class 3 component(s) do not conform with ASME Section XI requirements. This addition clarifies the existing TS and is consistent with the guidance provided by the Standard TS.

Add LCO 3.6.G.4

Add the following LCO:
"In RUN, STARTUP, or HOT SHUTDOWN MODE with Specification 3.6.G.2 or 3.6.G.3 not met:

a. perform an engineering evaluation to determine the effects of the component(s) condition for continued operation; and

b. determine that the component(s) remain acceptable for continued operation.

If the above requirements cannot be met, isolate the affected component(s) and follow the applicable system LCO."

The proposed LCO specifies actions in the event that a Class 1, 2, or 3 component(s) does not meet the structural integrity requirements of ASME Section XI while the reactor is in the RUN, STARTUP, or HOT SHUTDOWN MODE. The Standard TS do not provide any guidance for this condition. If this condition exists, an engineering evaluation will be performed to determine if the affected component(s) remain acceptable for continued operation. If the engineering evaluation justifies continued operation without any other actions (e.g., isolation of the affected component(s)), then no other action will be implemented. If the engineering evaluation determines that the component needs to be isolated, then such action will be initiated. Upon isolation of the component(s), the applicable system LCO will be followed.

3.6-8 Revise existing SR 4.6.G.1

The existing SR is being revised as follows:
"In-service inspection of ASME **Section XI** Code Class ~~I~~ **1**, Class ~~II~~ **2**, and Class ~~III~~ **3** components **and inservice testing of ASME Section XI Code Class 1, Class 2, and Class 3 pumps and valves** shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g)(6)(i)."

This surveillance is being editorially revised by adding the words "**Section XI**" after ASME to provide clarification and to specifically identify the applicable Code section even though it is implied in the existing SR. The hyphen in the existing "in-service" has been deleted. Also, the Code Class numbers are being changed from Roman numeral to digital as they are noted in the applicable ASME Code and the other TS.

The words "... **and inservice testing of ASME Section XI Code Class 1, Class 2, and Class 3 pumps and valves** ... ." are being added to this surveillance. This is an editorial change which relocated the subject wording from the existing SR 4.6.G.2, which is being deleted.

3.6-8 Delete existing SR 4.6.G.1.a

The following SR is being deleted:
"~~The second 10-year interval for the inservice inspection program described above commenced on November 1, 1985~~."

This information is being deleted from the SRs and is already incorporated into the Bases Section. The information provided here is not prudent to plant operation and does not serve to mitigate any accident, enhance plant operation, or ensure OPERABILITY of any equipment. This information does not constitute a SR and should not be located here.

3.6-8 Delete existing SR 4.6.G.2

The following SR is being deleted:
"~~In-Service testing of ASME Code Class I, Class II and Class III pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g) (6) (i)~~."

This SR is being deleted as discussed above for the revision of SR 4.6.G.1. Part of the first sentence "In-service testing of ASME Code Class I, Class II and Class III pumps and valves" is being incorporated into SR 4.6.G.1. The Code Class references are being changed as discussed above in SR 4.6.G.1. The rest of the SR is actually the same wording as in SR 4.6.G.1 and is therefore being deleted.

3.6-8 Delete existing SR 4.6.G.2.a

The following SR is being deleted:
"~~The second 10-year interval for the inservice testing program described above commenced on February 1, 1985~~."

This information is being deleted from the SRs and is already incorporated into the Bases Section. The information provided here is not prudent to plant operation and does not serve to mitigate any accident, enhance plant operation, or ensure OPERABILITY of any equipment. This information does not constitute a SR and should not be located here.

3.6-9 Revise existing SR 4.6.G.3 to new SR 4.6.G.2

The existing SR is being revised as follows:
"The **augmented** ~~inservice~~ inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods**,** ~~and~~ personnel, and sample expansion included in this ~~g~~ **G**eneric ~~l~~ **L**etter."

The only changes to this SR are editorial. The words "inservice" and "and" are being deleted. The DAEC program is identified as the augmented inspection program. The reason is that DAEC has requested and been granted relief from various parts of the original Inservice Inspection Program. This change makes the nomenclature more technically correct. The word "inservice" is not applicable and is therefore being deleted.

The word "and" is a superfluous word and is replaced by a "**,**".

The words "**G**eneric **L**etter" are in initial caps. This is an editorial change.

DAEC TS Section 3/4.6.H, Shock Suppressors (Snubbers)
The following are proposed changes to the DAEC Snubbers Section of the TS.

3.6-10 Revise existing LCO 3.6.H.1

The existing LCO is being revised as follows:
"During ~~all modes of operation, except Cold Shutdown and Refuel,~~ **RUN, STARTUP, and HOT SHUTDOWN MODES** all safety-related snubbers shall be ~~operable, except as noted in 3.6.H.2~~ **OPERABLE. In COLD SHUTDOWN and REFUELING MODES safety-related snubbers, located on those systems required to be OPERABLE, must be OPERABLE.**"

The existing LCO is being divided into those snubbers supporting equipment required to be OPERABLE during 1) **RUN, STARTUP, and HOT SHUTDOWN MODES** and 2) **COLD SHUTDOWN and REFUELING MODES.**

The words, "all modes of operation" are being replaced with the specific MODES "**RUN, STARTUP, and HOT SHUTDOWN MODES.**" The existing LCO requires that all snubbers be OPERABLE at all times. The LCO is being revised to ensure that those snubbers supporting systems required to be OPERABLE for a specific MODE of operation are in fact OPERABLE.

The word "**OPERABLE**" is a defined term in the DAEC TS and therefore is to be in all caps. This change is being made to provide consistency within this section and the entire TS.

The last sentence, "**In COLD SHUTDOWN and REFUELING MODES safety-related snubbers, located on those systems required to be OPERABLE, must be OPERABLE.**" is being added to specifically identify the requirements for those snubbers supporting equipment required to be OPERABLE during COLD SHUTDOWN and REFUELING MODES. This change is consistent as referenced above which divides this LCO into MODES of applicability. This change is also consistent with the guidance provided in the Standard TS.

3.6-10 Revise existing LCO 3.6.H.2

The existing LCO is being revised as follows:
"With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per ~~Specification~~ **Surveillance Requirement** 4.6.H.~~3~~ **4** on the supported component or declare the supported system

inoperable and follow the appropriate ~~Limiting Conditions For Operation~~ **LCO** for that system."

The following editorial changes are being made:

The word "Specification" is being replaced with **"Surveillance Requirement,"** which accurately identifies the referenced number 4.6.H.4. This is also to provide consistency throughout the DAEC TS Section 3.6.

The "Surveillance Requirement" number 4.6.H.3 is being revised to 4.6.H.**4**. This change is being made due to the addition of a new SR that precedes this 4.6.H.3

The words, "Limiting Conditions For Operation" are being replaced by the abbreviation **"LCO."** This is an industry accepted abbreviation and is consistent throughout the DAEC TS.

Add new Table 4.6.H-1

The subject Table is being added to support those proposed changes in SR 4.6.H.1 and is consistent with the NRC Generic Letter 90-09. Reference SR 4.6.H.1 below for additional discussion.

3.6-10 Revise existing SR 4.6.H

The existing SR is being revised as follows:
"Each safety-related snubber shall be demonstrated OPERABLE by performance of the following **augmented** ~~inservice~~ inspection program **and the Surveillance Requirements of 4.6.H.5 and 4.6.H.6."**

The word **"augmented"** is being added to provide clarification and specifically reflect its correct nomenclature at DAEC. DAEC, as other utilities, has been granted NRC relief from implementing the complete Inservice Inspection Program. Based on this relief, DAEC refers to the Inservice Inspection Program as an Augmented Inspection Program which is a more accurate statement.

The last part of the sentence, **"and the Surveillance Requirements of 4.6.H.5 and 4.6.H.6."** is being added, again for clarification and guidance. Providing these references in the SR eliminates the need to search out and identify additional SR requirements. Identifying

these SRs here will bring to the attention that other SRs contain requirements that need to be considered as well.

3.6-10 Replace existing SR 4.6.H.1

The existing SR is being replaced as follows:
"Visual Inspections

~~The inservice visual inspection of snubbers shall be performed in accordance with the following schedule:~~

~~Number of Snubbers Found Inoperable During Inspection or During Inspection Interval~~	~~Next Required Visual Inspection Interval~~
~~0~~	~~18 months ± 25%~~
~~1~~	~~12 months ± 25%~~
~~2~~	~~6 months ± 25%~~
~~3,4~~	~~124 days ± 25%~~
~~5,6,7~~	~~62 days ± 25%~~
~~≥,8~~	~~31 days ± 25%~~

~~The required inspection interval shall not be lengthened more than one step at a time.~~

Snubbers are categorized ~~in two groups, "accessible and inaccessible," based on their accessibility for inspection during reactor operation. These two groups will be inspected independently according to the above schedule.~~ **as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.6.H-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6.H-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment No.____ (NRC to assign no.)."**

The wording of the existing TS specify a schedule for snubber visual inspections based on the number of inoperable snubbers found during

the previous visual inspection. The schedules for visual inspections assume that refueling intervals will not exceed 18 months. Because the current schedule for snubber visual inspections is based only on the number of inoperable snubbers found during the previous visual inspection, irrespective of the size of the snubber population, licensees having a large number of snubbers find that the visual inspection schedule is excessively restrictive. Some licensees have spent significant resources and have subjected plant personnel to unnecessary radiological exposure to comply with the visual examination requirements.

To alleviate this situation, the NRC staff developed an alternate schedule for visual inspections that maintains the same confidence level as the existing schedule and generally allows the licensee to perform visual inspections and corrective actions during plant outages. This TS revision will reduce future occupational radiation exposure and is highly cost effective. This alternate inspection schedule is identified in Attachment B of Generic Letter 90-09 and is consistent with the Commission's Policy Statement on TS Improvements. Minor revisions to the wording of the proposed TS were made to maintain consistency with the current DAEC TS.

3.6-11 Revise existing SR 4.6.H.2

The existing SR is being revised as follows:
"Visual Inspection Acceptance Criteria

Visual inspection shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) ~~(for hydraulic snubbers) inspection of the hydraulic fluid reservoir and fluid connections, (3)~~ attachments to the foundation or supporting structure are secure, and (3) ~~(4) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen.~~ **fasteners for the attachment of the snubber to the component and to the snubber anchorage are secure.** Snubbers which appear inoperable as a result of visual inspection, ~~may be determined to be OPERABLE~~ **shall be classified as unacceptable and may be reclassified acceptable** for the purpose of establishing the next visual inspection interval, ~~providing~~ **provided** that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers**, irrespective of type,** that may be generically ~~or operationally~~ susceptible; and (2) the affected snubber is functionally tested in the as-found condition and

determined ~~to be~~ OPERABLE per ~~specifications~~ **Surveillance Requirements** 4.6.H.4 **5** or 4.6.H.~~5~~ **6.** ~~,as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined to be inoperable and cannot be considered OPERABLE via functional testing for the purpose of establishing the next visual inspection interval.~~ **All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the action requirements shall be met."**

The requirement, "(for hydraulic snubbers) inspection of the hydraulic fluid reservoir and fluid connections," is being deleted from this SR. This requirement is incorporated into DAEC snubber procedures. This requirement does not demonstrate OPERABILITY. Removal of this requirement is consistent with the guidance provided in the Standard TS.

The requirement, "in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen" is being replaced by the standard wording, **"fasteners for the attachment of the snubber to the component and to the snubber anchorage are secure."** DAEC currently removes snubbers in order to stroke test these snubbers. This practice will continue as related to snubber stroking. The proposed change ensures that the snubber is secured at both the anchorage and component. This will also ensure that the snubber is secured to perform its intended function.

The statement, "may be determined to be OPERABLE" is being replaced by **"shall be classified as unacceptable and may be reclassified acceptable."** This change provides additional guidance to reclassify an unacceptable snubber. This clarification is consistent with the guidance provided in the Standard TS.

The following editorial changes are being made to insure consistency with the Standard TS, to provide additional guidance, to make the sentence grammatically correct based on the proposed changes, and to provide clarity within the SR:

> The word "providing" is being replaced with the word **"provided."**

The words, "**irrespective of type**" are being added to provide clarity and is consistent with the Standard TS.

The words "or operationally" and "to be" are being deleted as superfluous wording based on the above proposed changes.

The word "specifications" is being replaced by "**Surveillance Requirements.**" This is to make the statement correct since the numbers being referenced are actually Surveillance Requirements.

The SR numbers "4.6.H.4" and "4.6.H.5" are being revised to "**4.6.H.5**" and "**4.6.H.6.**" This change is based on the addition of SR 4.6.H.3 which requires the proceeding numbers to be changed.

The words "as applicable." are being deleted as superfluous wording.

The statement, "However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined to be inoperable and cannot be considered OPERABLE via functional testing for the purpose of establishing the next visual inspection interval." is being deleted. According to the manufacturer, the purpose of the cover is to prevent dust or other impurities from collecting on the nozzle used to add hydraulic fluid. This cover does not keep the fluid from leaking out, thus making the snubber unable to perform its intended function. The manufacturer has stated that a damaged or missing dust cover does not render the snubber inoperable.

The following statement, "**All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the action requirements shall be met.**" is being added. This statement provides additional guidance which is not currently in the existing TS. This is also consistent with the guidance provided in the Standard TS.

<u>Add new SR 4.6.H.3</u>

Add the following SR:

"Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients, as determined from a review of operational data or a visual inspection of the systems, within 72 hours for accessible systems and 6 months for inaccessible systems following this determination. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range travel."

The current SRs do not require any inspection to be performed on the snubbers in the event that they have experienced an unexpected potentially damaging transient. To eliminate this concern, the proposed SR, **"Transient Event Inspection"** is being added which is consistent with the guidance provided by the Standard TS. In the event such a transient occurs, the proposed SR provides specific guidance as to what actions are required to ensure the potentially affected snubbers are capable of performing their intended function. The addition of this SR provides needed guidance, is consistent with the current industry practices, and enhances the overall snubber TS.

3.6-12 Revise the existing SR 4.6.H.3 to the new 4.6.H.4

The existing SR is being revised as follows:
"Functional Tests

Once per ~~operating cycle~~ **OPERATING CYCLE,** ~~*~~ a representative sample (10% of the total of safety-related of each type of snubber in use in the plant) shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of ~~specification~~ **Surveillance Requirements** 4.6.H.4 **5** or 4.6.H.~~5~~ **6**, an additional 5% of that type of snubber shall be functionally tested.

The representative sample selected for functional testing shall represent the various configurations, operating environments and range of sizes of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

~~1~~a. The first snubber away from each reactor vessel nozzle.

~~2~~b. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.).

~~3~~c. Snubbers within 10 feet of the discharge from a safety relief valve.

~~*This interval may be extended, on a one-time-only basis, for Cycle 8 operation until April 17, 1987. The subsequent test interval will begin with the actual Cycle 8/9 Refuel Outage test date.~~

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For any snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are restrained by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components restrained by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the component remains capable of meeting the designed service requirement."

The existing SR 4.6.H.3 has been revised and renumbered to SR 4.6.H.4. The SR number changed as a result of adding new SR 4.6.H.3 "Transient Event Inspection."

The words, **"OPERATING CYCLE"** are being revised to appear in all caps since this is a defined term in the DAEC TS. This is an editorial change to provide consistency throughout the DAEC TS.

The word "specification" is being replaced by "**Surveillance Requirements**." This is to make the statement correct since the numbers being referenced are actually Surveillance Requirements.

The SR numbers "4.6.H.4" and "4.6.H.5" are being revised to "**4.6.H.5**" and "**4.6.H.6**." This change is based on the addition of SR 4.6.H.3 which requires the proceeding numbers to be changed.

The numbers, "1, 2 and 3" have been replaced with "**a**, **b**, and **c**." This is an editorial change.

The footnote, "*This interval may be extended, on a one-time-only basis, for Cycle 8 operation until April 17, 1987. The subsequent test interval will begin with the actual Cycle 8/9 Refuel Outage test date.", is being deleted. This footnote discusses extending Cycle 8 operation with the subsequent interval actually beginning with the Cycle 8/9 Refuel Outage test date. DAEC has passed the subject Refueling Outages and therefore, the information contained in this footnote is no longer applicable and should be deleted. This footnote does not contain any information which is needed to support any surveillance testing of snubbers and does not contribute anything to the verification of snubber OPERABILITY.

3.6-13a/14 <u>Revise existing SR 4.6.H.4 to new SR 4.6.H.5</u>

The existing SR is being revised as follows:
"<u>Hydraulic Snubbers Functional Test Acceptance Criteria</u>

The hydraulic snubber functional test shall verify that:

~~1~~ **a**. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

~~2~~ **b**. Snubber bleed, or release rate is within the specified range in compression or tension. For snubbers specifically required ~~to~~ not **to** displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified."

The SR number has been changed from the existing SR 4.6.H.4 to the new SR 4.6.H.**5**. This change is being made due to the addition of the new SR 4.6.H.3. The contents of this SR are the same with the exception of the editorial change discussed below.

The editorial change made to this SR is that the word "**to**" is being moved to proceed the word "not". This is an editorial change and does not impact the intent of the SR.

The numbers, "1 and 2" have been replaced with "**a** and **b**." This is an editorial change.

3.6-14/15 Revise existing SR 4.6.H.5 to new SR 4.6.H.6

The SR number has been changed from the existing SR 4.6.H.5 to SR 4.6.H.**6**. This editorial change is a result of adding SR 4.6.H.3 which changes the numbering of any proceeding SRs.

The numbers, "1, 2, and 3" have been replaced with "**a**, **b**, and **c**." This is an editorial change.

Add new SR 4.6.H.7

The addition of a new SR is as follows:
"**Functional Testing of Repaired and Replaced Snubbers**

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit."

The existing SR does not contain any specific SR for functional testing of repaired or replaced snubbers. The addition of this SR eliminates that concern. The new SR requires that any snubber which fails the visual inspection or functional test acceptance criteria shall either be repaired or replaced. Replacement or repaired snubbers which might affect the existing functional test results shall be tested and meet the functional test criteria before installation.

This SR is consistent with current industry practices and with the guidance provided by the Standard TS. The addition of this SR will enhance the entire snubber TS.

Add new SR 4.6.H.8

The addition of the new SR is as follows:
"Snubber Service Life Replacement Program

The service life of all snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.12."

This SR is being added to provide specific guidance for the snubber service life replacement program. The existing SR 4.6.H.6 contains some of the information presented in this proposed SR such as record storage. The additional requirements specified in this SR are to enhance the repair and replacement program. These enhancements will further assure that the maximum service life will not be exceeded during the period the snubber is required to be OPERABLE. The addition of this SR provides clarification and specific guidance as contained in the Standard TS.

3.6-15 Delete existing SR 4.6.H.6

This existing SR is being deleted as follows:
~~"Snubber Service Life Monitoring~~

~~A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.13.~~

~~Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded and will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records."~~

This entire SR is being deleted. The majority of the SR is incorporated into the proposed SRs 4.6.H.7 and 4.6.H.8. The requirement for the maintenance of a record of each snubber in accordance with Specification 6.10.2.12 is retained. The only change is that Specification 6.10.2.13 is the wrong reference number. The correct specification number is Specification 6.10.2.12.

The second paragraph of existing SR 4.6.H.6 requires that the records for each snubber be reviewed to ensure that the service life of the snubber is not exceeded before the next inspection interval. Again, this requirement is incorporated into the new proposed SR 4.6.H.8. The new SR requires that the service life of all snubbers be monitored to ensure that the service life is not exceeded during surveillance intervals.

The existing SR 4.6.H.6 also requires that if the indicated service life will be exceeded before the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. The new SR 4.6.H.8 requires that the expected service life for various seals, springs, and other critical parts shall be extended or shortened based on monitored test results. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE.

The last requirement in the existing SR 4.6.H.6 states that the reevaluation, replacement or reconditioning shall be indicated in the records. The new SR 4.6.H.8 incorporates the intent of this requirement in that it states that the parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.12.

TECHNICAL SPECIFICATIONS
LIST OF TABLES (Continued)

41. IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known not to interfere with the operation of the leakage detection systems.

42. TOTAL LEAKAGE

TOTAL LEAKAGE is the sum of IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE.

43. UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

44. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram(ml), which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

TABLE 1.0-1

OPERATING MODES

OPERATING MODE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. RUN/POWER OPERATION	Run	NA
2. STARTUP	Startup/Hot Standby or Refuel[a]	NA
3. HOT SHUTDOWN[a]	Shutdown[c][d]	> 212°F
4. COLD SHUTDOWN[a]	Shutdown[c][d][e]	≤ 212°F
5. REFUELING[b]	Shutdown or Refuel[c][f]	NA

(a) Fuel in the reactor vessel with the reactor vessel head closure bolts fully tensioned.

(b) Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

(c) The reactor mode switch may be placed in the Run, Startup/Hot Standby or Refuel position to test the switch interlock functions and related instrumentation provided that the control rods are verified to remain fully inserted by a second licensed operator.

(d) The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

(e) The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.A.

(f) The reactor mode switch may be placed in the Startup position for demonstration of shutdown margin per Specification 4.3.A.1.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the Reactor Coolant System.

Objective:

To assure the integrity and safe operation of the Reactor Coolant System.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup and cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

2. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Curve C of Figure 3.6-1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Figure 3.6-1.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the Reactor Coolant System.

Objective:

To determine the condition of the Reactor Coolant System and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the temperatures at the following locations shall be recorded at least every 15 minutes until 3 consecutive readings at each location are within 5°F:

   a. reactor vessel shell adjacent to shell flange.

   b. reactor vessel bottom drain.

   c. recirculation loops A and B.

   d. reactor vessel bottom head temperature.

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every 15 minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER as required by 10CFR50, Appendix H. The results of these fluence determinations shall be used to update Figure 3.6-1.
3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the head flange and the shell adjacent to the head flange is greater than or equal to 74°F.	3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.
4. With any of the above limits exceeded: a. restore the temperature and/or pressure to within the limits within 30 minutes, and b. within 72 hours perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System, and c. determine that the Reactor Coolant System remains acceptable for continued operation; or d. if the above requirements cannot be met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	
5. When in RUN, STARTUP, HOT SHUTDOWN, COLD SHUTDOWN, or REFUELING MODE, the following LCOs apply to the idle recirculation loop startup: a. A reactor recirculation pump shall not be started unless the reactor coolant temperature differential between the dome and the bottom head drain is less than or equal to 145°F.	4. Prior to starting a recirculation pump, the following reactor coolant temperatures shall be within limits and recorded: a. differential between the dome and the bottom head drain.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
b. When only one recirculation loop is idle, the recirculation pump shall not be started unless the temperature differential of the reactor coolant between the idle and operating recirculation loops is less than or equal to 50°F.	b. differential between the recirculation loops.
B. Coolant Chemistry	B. Coolant Chemistry
1.a. With the reactor critical, the specific activity of the primary coolant shall be less than or equal to 1.2 μCi/ml DOSE EQUIVALENT I-131.	1.a. The specific activity of the reactor coolant shall be demonstrated to be within limits by performance of the sampling and analysis program of Table 4.6.B.1-1.
b. When in the RUN, STARTUP, or HOT SHUTDOWN MODES, the specific activity of the primary coolant can be greater than 1.2 μCi/ml DOSE EQUIVALENT I-131 for a maximum of 48 hours, provided that the DOSE EQUIVALENT I-131 activity does not exceed 12.0 μCi/ml during this time. The reactor shall not be operated more than 5 percent of its yearly power operation under this exception for equilibrium activity limits.	b. Whenever the DOSE EQUIVALENT I-131 exceeds 0.6 μCi/ml, notify the USNRC as specified by 6.11.1.h.
c. If the specific activity of the primary coolant is greater than 12.0 μCi/ml DOSE EQUIVALENT I-131, the reactor shall be shutdown, and the Main Steam Line Isolation Valves shall be closed immediately.	
2. At all times the chemistry of the Reactor Coolant System shall be maintained within the limits specified in Table 3.6.B.2-1.	2. The reactor coolant shall be determined to be within the specified chemistry limits by:
a. In RUN MODE:	a. Measurement prior to pressurizing the reactor during each startup, if not performed within the previous 72 hours.
1) With any limit in Table 3.6.B.2-1 exceeded for more than: a) 720 hours per year, or b) 72 continuous hours, be in at least STARTUP within 6 hours.	b. Obtain and analyze a sample of the reactor coolant at least once every 72 hours for chlorides and conductivity.*
	* Not applicable with no fuel in the reactor vessel.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2) With the conductivity exceeding 10.0 μmho/cm at 25°C or chloride concentration exceeding 500 ppb, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	c. Obtain and analyze a sample of the reactor coolant for chlorides at least once every 8 hours whenever conductivity is greater than the limit specified in Table 3.6.B.2-1.
	d. Obtain and analyze a sample of the reactor coolant for pH at least once every 8 hours whenever conductivity is greater than the limit specified in Table 3.6.B.2-1.
3) Continuously record the conductivity of the reactor coolant. With no continuous recording conductivity monitor OPERABLE, install a temporary in-line conductivity monitor within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	e. With no continuous recording conductivity monitor OPERABLE, obtain an in-line conductivity measurement at least once per 4 hours when in RUN, STARTUP, or HOT SHUTDOWN MODES and 24 hours at all other times.
	f. Perform a CHANNEL CHECK of the continuous conductivity monitor at least once per 7 days.
b. In STARTUP and HOT SHUTDOWN:	
1) With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.B.2-1 for more than 48 continuous hours, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	
c. In COLD SHUTDOWN and REFUEL:	
1) With the conductivity or pH exceeding the limit specified in Table 3.6.B.2-1, restore the conductivity and pH to within the limit within 72 hours.	
2) With chloride concentration exceeding the limit specified in Table 3.6.B.2-1, restore the chloride concentration to within the limit within 24 hours.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
d. If the conditions in Specification 3.6.B.2.c.1 or 3.6.B.2.c.2 above cannot be met: 1) perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System, and 2) determine that the structural integrity of the Reactor Coolant System remains acceptable for continued operation prior to leaving COLD SHUTDOWN.	

TABLE 3.6.B.2-1
REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS

MODES	CHLORIDES	CONDUCTIVITY μMHOS/CM @25°C	PH
RUN	≤ 200 ppb	≤ 1.0	5.6 ≤ PH ≤ 8.6
STARTUP/HOT SHUTDOWN	≤ 100 ppb	≤ 2.0	5.6 ≤ PH ≤ 8.6
COLD SHUTDOWN/ REFUELING*	≤ 100 ppb	≤ 5.0	4.6 ≤ PH

* Not applicable with no fuel in the reactor vessel

TABLE 4.6.B.1-1
PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE OF ANALYSIS FREQUENCY	OPERATING CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Isotopic Analysis of Filtrate from a 0.45 μ filter and gross iodine activity and DOSE EQUIVALENT I-131 and I-131 and I-133 determination	At least once per 72 hours	RUN, STARTUP, and HOT SHUTDOWN
2. Isotopic Analysis including I-131, I-132, I-133 and I-135	At least monthly	RUN
3. Isotopic Analysis for gross iodine and DOSE EQUIVALENT I-131	a) Within 24 hours prior to startup b) At least once per 4 hours, if the DOSE EQUIVALENT I-131 exceeds the limit as required in TS 3.6.B.1.b	RUN*, STARTUP*, HOT SHUTDOWN*, or COLD SHUTDOWN*

* Until specific activity of the primary coolant system is restored to within its limits.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
C. Coolant Leakage	C. Coolant Leakage
1. When in RUN, STARTUP, or HOT SHUTDOWN MODE, the Reactor Coolant System leakage into the drywell shall be limited to: a. ≤ 5 gpm UNIDENTIFIED LEAKAGE. b. ≤ 2 gpm increase in UNIDENTIFIED LEAKAGE within a 24 hour period. c. ≤ 25 gpm TOTAL LEAKAGE.	1. Reactor Coolant System leakage shall be checked by the Sump System and recorded at least once every 8 hours.
2. With the conditions in Specifications 3.6.C.1.a, b, or c above not met, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	
3. When in RUN, STARTUP, or HOT SHUTDOWN MODE, the Sump System shall be OPERABLE as defined in Table 3.2-E.	2. Verify Sump System OPERABILITY as specified in Table 4.2-E.
4. With the Sump System inoperable, immediately verify the Air Sampling System is OPERABLE and restore the Sump System to OPERABLE status within the next 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	3. Verify Air Sampling System OPERABILITY as specified in Table 4.2-E. The Air Sampling System shall be checked and recorded at least once every 8 hours.
5. With both the Sump System and the Air Sampling System inoperable, restore one of the systems to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
D. Safety and Relief Valves	D. Safety and Relief Valves
1. When in RUN, STARTUP, or HOT SHUTDOWN MODE, both safety valves and the safety modes of all relief valves* shall be OPERABLE, except as specified in Specification 3.6.D.2.	1. Once per OPERATING CYCLE, at least one safety valve and 3 relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested. The safety and relief valves shall be rotated, at least once per 40 months, such that both safety and 6 relief valves are removed, set pressure tested and reinstalled or replaced with spares. Any spare that is installed must have been set pressure tested within the previous 40 months. The setpoint of the safety valves shall be as specified in Specification 2.2.
2.a With the safety valve function of one relief valve inoperable, restore the inoperable safety valve function to OPERABLE status within thirty days. b. With the safety valve function of two relief valves inoperable, restore the inoperable safety valve function to OPERABLE status within seven days.	2. At least one of the relief valves shall be disassembled and inspected once per OPERATING CYCLE.
3. If Specification 3.6.D.1 or 3.6.D.2 is not met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	3. With the reactor pressure $\geq$ 100 psig and turbine bypass flow to the main condenser, each relief valve shall be manually opened and verified open by turbine bypass valve position decrease, pressure switches and thermocouple readings downstream of the relief valve to indicate steam flow from the valve once per OPERATING CYCLE. 4. The relief valve setpoints for the Low-Low Set function shall be as specified in Section 2.2.1.c. Instrumentation and system logic shall be functionally tested, calibrated, and checked as specified in Table 4.2-B.
* SRVs which perform an ADS function must also satisfy the OPERABILITY requirements of Specification 3.5.F, Core and Containment Cooling Systems.	

LIMITING CONDITIONS FOR OPERATION

E. Jet Pumps

1. Whenever the reactor is in the RUN or STARTUP MODE, all jet pumps shall be OPERABLE.

a. If one or more jet pumps do not meet the Surveillance Requirements of 4.6.E.2 with:

1) the recirculation pump speed less than 60% of rated, continue to monitor the jet pump(s) performance per Surveillance Requirement 4.6.E.2 daily until the evaluation can be performed at pump speed greater than 60% of rated.

2) the recirculation pump speed greater than or equal to 60% of rated, evaluate the reason for the deviation. If the evaluation verifies the jet pump(s) to be inoperable, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

E. Jet Pumps

1. Jet pump OPERABILITY shall be verified daily, following startup of a recirculation pump and after any unexplained changes in either core flow, jet pump loop flow, recirculation loop flow, or core plate differential pressure (ΔP), by recording the jet pump diffuser to lower plenum ΔP's, recirculation pump flows, recirculation pump speeds, and jet pump loop flows and verifying that:

a. the recirculation pump flow to pump speed ratio does not vary from the normal expected operating range by more than 5%, and

b. the jet pump loop flow to recirculation pump speed ratio does not vary from the normal expected operating range by more than 5%.

c. if the Surveillance Requirements of 4.6.E.1.a or 4.6.E.1.b are not met, perform the Surveillance Requirements of 4.6.E.2 within 24 hours.

2. Record the individual jet pump ΔP's and verify that the individual jet pump ΔP percent deviation from the average loop ΔP does not vary from its normal expected operating range by more than 20%.

3. The Surveillance Requirements of 4.6.E.1 and 4.6.E.2 do not apply to the idle recirculation loop and associated jet pumps when in SLO.

4. Following each REFUEL OUTAGE, as soon as practical after reaching 60% of rated pump speed, update the baseline data used to perform the above evaluations. Baseline data for SLO shall be updated as soon as practical after entering SLO.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
F. Jet Pump Flow Mismatch	F. Jet Pump Flow Mismatch
1. With core power greater than or equal to 80% RATED POWER with both recirculation pumps at steady state operation, the speed of the faster pump may not exceed 122% of the speed of the slower pump.	1. Recirculation pump speed mismatch shall be verified at least once per day.
2. With core power less than 80% RATED POWER with both recirculation pumps at steady state operation, the speed of the faster pump may not exceed 135% of the speed of the slower pump.	
3. With the recirculation pump speeds different by more than the specified limits:	
a. restore the recirculation pump speeds to within the specified limit within 2 hours, or	
b. one recirculation pump shall be tripped. See Specification 3.3.F.4 for SLO requirements.	2. See Surveillance Requirement 4.3.F.4 for SLO requirements.
G. Structural Integrity	G. Structural Integrity
1. At all times, the structural integrity of the ASME Section XI Code Class 1, 2, and 3 components shall be maintained in accordance with Surveillance Requirement 4.6.G.1.	1. Inservice inspection of ASME Section XI Code Class 1, Class 2, and Class 3 components and inservice testing of ASME Section XI Code Class 1, Class 2, and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g)(6)(i).
2. With the structural integrity of any ASME Section XI Code Class 1 or Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 212°F.	2. The augmented inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this Generic Letter.
3. With the structural integrity of any ASME Section XI Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
4. In RUN, STARTUP, or HOT SHUTDOWN MODE with Specification 3.6.G.2 or 3.6.G.3 not met: a. perform an engineering evaluation to determine the effects of the component(s) condition for continued operation; and b. determine that the component(s) remain acceptable for continued operation. If the above requirements cannot be met, isolate the affected component(s) and follow the applicable system LCO.	
H. Shock Suppressors (Snubbers)	H. Shock Suppressors (Snubbers)
1. During RUN, STARTUP, and HOT SHUTDOWN MODES all safety-related snubbers shall be OPERABLE. In COLD SHUTDOWN and REFUELING MODES safety-related snubbers, located on those systems required to be OPERABLE, must be OPERABLE.	Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inspection program and the Surveillance Requirements of 4.6.H.5 and 4.6.H.6.
2. With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Surveillance Requirement 4.6.H.4 on the supported component or declare the supported system inoperable and follow the appropriate LCO for that system.	1. Visual Inspections Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.6.H-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6.H-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment No. (NRC to assign no.).

TABLE 4.6.H-1
SNUBBER VISUAL INSPECTION INTERVAL

NUMBER OF UNACCEPTABLE SNUBBERS

Population or Category (Notes 1 and 2)	Column A Extend Interval (Note 3)	Column B Repeat Interval (Note 4)	Column C Reduce Interval (Note 5)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

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2. Visual Inspection Acceptance Criteria

Visual inspection shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspection, shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers, irrespective of type, that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Surveillance Requirements 4.6.H.5 or 4.6.H.6. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the action requirements shall be met.

3. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients, as determined from a review of operational data or a visual inspection of the systems, within 72 hours for accessible systems and 6 months for inaccessible systems following this determination. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; (2)

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evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range travel.

4. Functional Tests

Once per OPERATING CYCLE, a representative sample (10% of the total of safety-related of each type of snubber in use in the plant) shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Surveillance Requirements 4.6.H.5 or 4.6.H.6, an additional 5% of that type of snubber shall be functionally tested.

The representative sample selected for functional testing shall represent the various configurations, operating environments and range of sizes of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

a. The first snubber away from each reactor vessel nozzle.

b. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.).

c. Snubbers within 10 feet of the discharge from a safety relief valve.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be

LIMITING CONDITIONS FOR OPERATION

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independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For any snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are restrained by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components restrained by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the component remains capable of meeting the designed service requirement.

5. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

a. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

b. Snubber bleed, or release rate is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

6. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

a. The drag force of any snubber in tension and compression is less than the specified maximum drag force.

b. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

c. Snubber release rate, where required, is within the

LIMITING CONDITIONS FOR OPERATION

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specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

7. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit.

8. Snubber Service Life Replacement Program

The service life of all snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.12.

3.6.A and 4.6.A BASES:

Thermal and Pressurization Limitations

The thermal limitations for the reactor vessel meet the requirements of 10 CFR 50, Appendix G, revised May 1983. (Ref. 3)

The allowable rate of heatup and cooldown for the reactor vessel contained fluid is 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time period for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation. In the event that the 100°F per hour heatup is exceeded, the plant will be brought back within limits within 30 minutes. In addition, within 72 hours an engineering evaluation is to be performed to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System. During this period, the plant will be placed in a stable condition until the evaluation is completed. During a heatup this can mean that startup activities may continue until the operator reaches a stable condition. If in a cooldown, and the temperature rate of change exceeds the 100°F per hour limit, the reactor will be brought back to within limits with cooldown continuing.

Specific stress analyses for the reactor vessel materials were made based on a heatup and cooldown rate of 100°F/hour applied continuously over a temperature range of 100°F to 546°F. Calculated stresses were found to be within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

Chicago Bridge and Iron Company performed detailed stress analyses as shown in the Updated FSAR Appendix 5A, "Site Assembly of the Reactor Vessel." The analyses include more severe thermal conditions than those which would be encountered during heatup and cooldown operations.

The permissible flange to adjacent shell temperature differential of 145°F is the maximum calculated for 100°F per hour heatup and cooldown rate applied continuously over a 100°F to 550°F range. The differential is due to the sluggish temperature response to the flange metal and its value decreases for any lower heatup rate or the same rate applied over a narrower range.

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 which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The Reactor Coolant System is a primary barrier against the release of fission products to the environment. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

The operating limits in Figure 3.6-1 are derived in accordance with 10CFR50, Appendix G, May, 1983 and Appendix G of the ASME Code. Conditions in three regions influence the curves: the closure flange region, the non-beltline region which includes most nozzles and discontinuities, and the beltline region which is irradiated with fluence above 10^{17} n/cm^2 during the vessel operating life. Irradiation has caused an increase in the nil-ductility temperature (RT_{NDT}) of the beltline materials, to the point where the beltline region impacts the pressure-temperature limits for the vessel. For Figure 3.6-1, effective to 16 EFPY, the beltline which has an RT_{NDT} of 40°F is limiting at higher pressures. The non-beltline regions which generally experience higher stresses at nozzles and discontinuities are limiting at lower pressures. The limiting RT_{NDT} of 58°F for the Standby Liquid Control Nozzle (N10) is the highest RT_{NDT} of any component in the non-beltline region. The closure flange region, with RT_{NDT} = 14°F, has a bolt preload and minimum operating temperature of 74°F. This exceeds original requirements of the ASME Code (Winter 1967 Addendum) and provides extra margin relative to current ASME

Code requirements.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The first capsule was removed after fuel cycle 7, at 6 effective full power years. The neutron flux wires tested were used to determine the end-of-life fluence at the 1/4 T depth in the vessel wall of 3.6×10^{18} n/cm^2. Test specimens of the reactor vessel base, weld and heat affected zone material were installed in the reactor vessel adjacent to the vessel wall at the core midplane level at the start of operation. The next surveillance capsule shall be withdrawn at 15 effective full power years and tested in accordance with 10CFR50, Appendix H. Samples from surveillance capsule 1 at vessel azimuth 288° were withdrawn at 6 effective full power years and tested in accordance with 10CFR50, Appendix H. Neutron flux wires installed in the surveillance capsule were tested to experimentally determine the flux and fluence at the 1/4 T depth of the beltline shell thickness, used to determine the NDTT shift. The next surveillance capsule shall be withdrawn at 15 effective full power years. Irradiated and unirradiated Charpy specimens were tested. Since the test showed that the limiting beltline material initial RT_{NDT} and the RT_{NDT} shift are the same as those previously predicted, there was no need to change the curves of Figure 3.6-1 based on Surveillance Materials Testing. However, an adjusted reference temperature, based on the fluence, nickel content and copper content of the material in question, can be predicted using the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure-temperature curves of Figure 3.6-1 includes predicted adjustments for this shift in RT_{NDT} at the end of 16 EFPY. New curves for Figure 3.6-1 will be submitted prior to reaching 16 EFPY. Future shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185 and 10CFR50, Appendix H, irradiated reactor vessel materials installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material nil ductility transition temperature shift. Operating limits of Figure 3.6-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2. In the event that the pressure/temperature limits are exceeded, they are to be restored to within the limits within 30 minutes. Once restored, an evaluation is to be performed in order to determine if operation can continue. The purpose of this evaluation is to verify reactor coolant pressure boundary integrity is maintained and was not jeopardized.

As described in paragraph 4.2.5 of the Safety Analysis report, 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 transients are compared to allowable stress limits. In order to prevent undue stress on the vessel nozzles and bottom head region, the idle recirculation loop temperature shall be within 50°F of the operating loop temperature prior to startup of an idle recirculation pump.

3.6.B & 4.6.B BASES:

Coolant Chemistry

The basis for the equilibrium coolant iodine activity limit is a computed dose to the thyroid of 30 rem at the exclusion distance during the 2-hour period following a steam line break. This dose is computed with the conservative assumption of a release of 140,000 lbs. of coolant prior to closure of the Main Steam Line Isolation Valves and Regulatory Guide 1.5 Meteorology.

The maximum activity limit during a short term transient is established from consideration of a maximum iodine inhalation dose less than 300 rem. The probability of a steam line break accident coincident with an iodine concentration transient is significantly lower than that of the accident alone, since operation of the reactor with iodine levels above the equilibrium value is limited to 5 percent of total operation.

General Electric review of daily reactor water iodine concentrations at several sites indicates that the iodine transients during power generation are less than a factor of ten. Sampling frequencies have been established that vary with the iodine concentration in order to assure that the maximum coolant iodine concentrations are not exceeded.

Materials in the primary system are primarily stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel. According to test data, allowable chloride concentrations could be set several orders of magnitude above the established limit at the oxygen concentration (200 - 300 ppb) experienced during power operation without causing significant failures. Zircaloy does not exhibit similar stress corrosion failures. However, there are some conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 200 - 300 ppb, such as refueling, reactor startup and HOT STANDBY. During these periods, a limit of 100 ppb has been established to assure that permissible chloride-oxygen combinations are not exceeded. Boiling occurs at higher steaming rates causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels and assuring that the chloride-oxygen content is not such as would tend to induce stress corrosion cracking.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors conductivities are in fact high due to purposeful addition of additives. In the case of BWR's however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-normal condition include operation of the Reactor Water Cleanup (RWCU) System, reducing the input of impurities and placing the reactor in the COLD SHUTDOWN condition. The major benefit of COLD SHUTDOWN is to reduce the temperature dependent corrosion rates and provide time for the RWCU System to re-establish the purity of the reactor coolant. During some periods of operation, conductivity or chloride concentration may exceed 5.0 μmo/cm or 200 ppb respectively because of the initial evolution of gases, the initial addition of dissolved metals, or the breaking out of chlorides entrapped in the system. The total time during which the conductivity or

chloride concentration may exceed the specified limit must be limited to 2 weeks/year or less to prevent stress corrosion cracking.

At DAEC, conductivity is continuously monitored at the Reactor Water Cleanup System, between the hot well and the demineralizer beds, and at the outlet of the demineralizer beds. Any of these monitors are considered to fulfill the requirement of continuously monitoring the Reactor Coolant System. In the event that the conductivity cannot be continuously monitored, a temporary in-line monitor is to be installed.

The iodine radioactivity will be monitored by reactor water sample analysis. The total iodine activity would not be expected to change over a period of 1 week. In addition, the trend of the offgas stack release rate, which is continuously monitored, is an indication of the trend of the iodine activity in the reactor coolant. Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, the capability to detect gross fuel element failures is inherent in the radiation monitors in the Offgas System and on the main steam lines.

The conductivity of the reactor coolant is continuously monitored. Conductivity instrumentation will be checked every 3 days by instream measurements with an independent conductivity monitor to assure accurate readings. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses to determine major contributors to activity can be performed by a gamma scan.

3.6.C & 4.6.C BASES:

Coolant Leakage

Allowable leakage rates of coolant from the Reactor Coolant System have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to make up coolant system leakage in the event of loss of offsite a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for UNIDENTIFIED LEAKAGE, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable UNIDENTIFIED LEAKAGE greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

IDENTIFIED and UNIDENTIFIED LEAKAGE are defined in the DAEC Technical Specifications Section 1.0, "Definitions" and in the Updated FSAR, Section 5.2.5.2.2. TOTAL LEAKAGE is defined as the sum of IDENTIFIED and UNIDENTIFIED LEAKAGE. IDENTIFIED LEAKAGE is that leakage entering the drywell equipment drain sump. Identifiable leakage into the drywell equipment drain sump is composed of normal seal and valve packing leakage and does not represent a safety consideration so long as the leakage is small compared to the available reactor coolant makeup capacity.

Unidentifiable leakage is composed of all leakage from the reactor primary system that is not defined as identifiable leakage. This unidentifiable leakage is collected in the drywell floor drain sump.

In the event that UNIDENTIFIED LEAKAGE has been identified, it may be reclassified as IDENTIFIED LEAKAGE with the applicable IDENTIFIED LEAKAGE limit now applying.

A total allowable leakage of 25 gpm will be the sum of the UNIDENTIFIED LEAKAGE and the IDENTIFIED LEAKAGE. The drywell floor drain sump and the equipment drain sump both have two pumps with each pump having a capacity of 50 gpm. Removal of the allowable TOTAL LEAKAGE from either of these sumps can be accomplished with margin.

DAEC surveillance procedures require IDENTIFIED and UNIDENTIFIED LEAKAGE to be determined at approximately 4 hour intervals. Should leakage exceed the allowed limits, control room alarms actuated by the equipment drain sump and floor drain sump pump timers are prcvided to indicate this condition, thus, continuous leakage detection capability is provided by design.

The requirement that an increase in UNIDENTIFIED LEAKAGE shall not exceed 2 gpm in a 24 hour period is based on maintaining the ability to detect small leaks in a reasonably short time such that corrective action can be initiated. However, during reactor startup and ascension to normal operating pressure, leakage should be closely monitored until normal operating pressure is achieved and a "baseline" leakage rate can be established to which any leakage increase can be compared. An UNIDENTIFIED LEAKAGE rate increase of >2 gpm over a 24 hour period is an indication of a potential flaw in the reactor

pressure boundary. Even though the >2 gpm UNIDENTIFIED LEAKAGE does not exceed the ≤ 5 gpm UNIDENTIFIED LEAKAGE limit, certain components must be determined not to be the source of the increased leakage.

The primary containment atmosphere radioactivity detector provides a sensitive and rapid indication of increased nuclear system leakage. The primary containment environment is continuously sampled from one of three locations which are chosen to provide both a representative gas mixture and an indication of the location of the leakage.

The sample air undergoes three separate processes in which the radioactive noble gas, halogen, and particulate contents are determined. This system is thus a three channel monitoring system. The processed air is returned to the drywell.

The primary containment atmosphere radioactivity detector serves as a sensitive, reliable backup to the other methods of leak detection. It is anticipated that the particulate detector will be the primary indication of leakage, with the halogen and noble gas detectors serving as indication of the primary containment environment if primary containment venting is required. These detectors in conjunction with an isotopic analysis can be used to indicate whether the detected leak is from a steam or water system. This system is not capable of accurately quantifying coolant leakage rates. Because the Air Sampling System is not capable of determining leak rate, it is considered a backup system to the Sump System, and no LCO is associated with it. It is intended to be a compensatory measure used when the Sump System is inoperable.

3.6.D & 4.6.D BASES:

Safety and Relief Valves

The pressure relief system has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet power generation design basis #1 of Section 5.4.13.1 of the Updated FSAR, which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME Code requirements is presented in Subsection 5.4.13 of the Updated FSAR and is reverified in individual reload analyses.

Six relief valves and two safety valves are installed. The analysis of the worst overpressure transient, (3-second closure of all Main Steam Line Isolation Valves) neglecting the direct scram (valve position scram) results in a peak vessel pressure less than the Code allowable overpressure limit of 1375 psig if a flux scram is assumed.

The relief valve setpoints given in Section 2.2.1.B have been optimized to maximize the simmer margin, i.e., the difference between the normal operating pressure and the lowest relief valve setpoint. The Reference 2 analysis shows that the six relief valves assure margin below the setting of the safety valves such that the safety valves would not be expected to open during any normal operating transient.* This analysis verifies that the peak system pressure during such an event is limited to greater than the 60 psi design margin to the lowest spring safety valve setpoint.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per OPERATING CYCLE is adequate to detect failures or deteriorations. The relief and safety valves are benchtested every second OPERATING CYCLE to ensure that their setpoints are within the ± 1 percent tolerance. Additionally, once per OPERATING CYCLE, each relief valve is tested manually with reactor pressure above 100 psig and with turbine bypass flow to the main condenser to demonstrate its ability to pass steam. By observation of the change in position of the turbine bypass valve, the relief valve operation is verified.

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The surveillance requires that at least once per OPERATING CYCLE at least one safety valve and 3 relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested. For the most part, these valves will be set pressure tested and stored in accordance with the manufacturer's recommendations. There may be conditions where DAEC may not be notified by the manufacturer of new storage requirements or DAEC may take exception with the requirements. In these isolated cases, DAEC and the manufacturer will come to resolution on an acceptable position.

*A normal operating transient is defined as an event whose probability of occurrence is greater than once per 40 years, e.g., Turbine Trip with Bypass, MSIV closure with direct scram.

The low-low set (LLS) function provides automatic relief mode setpoints on the two non-ADS safety/relief valves (SRV's). The LLS function lowers the opening and closing setpoints after any SRV has opened at its normal steam pilot setpoint when a concurrent high reactor vessel steam dome pressure scram signal is present. The purpose of the LLS is to mitigate the induced high frequency loads on the containment and thrust loads on the SRV discharge lines. The LLS function increases the amount of reactor depressurization during an SRV blowdown because the lowered LLS setpoints keep the two LLS SRV's open for a longer time. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the LLS function such that failure of any one LLS valve to open or close at its reduced setpoint does not violate the design basis. (Ref. 1)

3.6.E & 4.6.E BASES:

Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser increases the cross sectional flow area for blowdown following the postulated design basis double-ended recirculation line break, i.e., the design basis LOCA. Therefore, if such a failure occurs, repairs must be made to assure the DAEC LOCA evaluations remain valid, and the plant does not operate outside its analyzed envelope.

The following factors form the basis for the surveillance requirements:

a. Recirculation Pump Flow/Speed Ratio: the pump operating characteristic is determined by the flow resistance from the loop suction through the jet pump nozzle. Since this resistance is essentially independent of core power, the flow is linearly proportional to pump speed, making their ratio a constant (flow/RPM is constant). A decrease in the ratio indicates a plug, flow restriction, or loss in pump hydraulic performance. An increase indicates a leak or new flow path between the recirculation pump discharge and jet pump nozzle.

b. Jet Pump Loop Flow/Recirculation Pump Speed Ratio: this relationship is an indication of overall system performance.

c. Jet Pump Differential Pressure Relationships: if a potential problem is indicated, the individual jet pump differential pressures are used to determine if a problem exists since this is the most sensitive indicator of significant jet pump performance degradation.

However, these tests are not very accurate below 60% of rated recirculation pump speed due to the instrument accuracy and the significant influence of natural circulation at core flows less than 50% of rated. Therefore, anomalous readings should be evaluated at higher pump speeds before declaring a jet pump inoperable.

After CORE ALTERATIONS, particularly when new fuel designs are loaded into the core, the established relationships for monitoring recirculation system performance may be affected. Hence the requirement to re-evaluate the data base after each refuel outage to determine if the baseline data for normal expected operation range remain valid. As stated above, the data is not very reliable below 60% of rated pump speed; thus, the re-evaluation of the data base should be performed after reaching 60% pump speed.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the sixteen individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow in the case of a failed jet pump. Thus the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing jet pump.* Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operating map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle-riser system failure.

*Note: In the case of SLO, when the recirculation pump is tripped, the flow through the inactive jet pumps is subtracted from the total jet pump flow, yielding the correct value for the total core flow.

3.6.F & 4.6.F BASES:

Jet Pump Flow Mismatch

The LPCI loop selection logic has been previously described in the Updated FSAR Section 7.3.1.1.2.4. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could be expected to function at a speed differential up to 14% of their average speed. Below 80% power the loop select logic would be expected to function at a speed differential up to 20% of their average speed. This specification provides margin because the limits are set at ±10% and ±15% of the average speed for the above and below 80% power cases, respectively. If the reactor is operating on one recirculation pump, the loop select logic trips that pump before making the loop selection.

3.6.G & 4.6.G BASES:

Structural Integrity

A pre-service inspection of Nuclear Class I Components was conducted to assure freedom from defects greater than code allowance; in addition, this served as a reference base for future inspections. Prior to operation, the Reactor Coolant System as described in Article IS-120 of Section XI of the ASME Boiler and Pressure Vessel Code was inspected to provide assurance that the system was free of gross defects. In addition, the facility was designed such that gross defects should not occur throughout plant life. The pre-service inspection program was based on the 1970 Section XI of the ASME Code for in-service inspection. This inspection plan was designed to reveal problem areas (should they occur) before a leak in the coolant system could develop. The program was established to provide reasonable assurance that no LOCA would occur at the DAEC as a result of leakage or breach of pressure-containing components and piping of the Reactor Coolant System, portions of the ECCS, and portions of the reactor coolant associated auxiliary systems.

A pre-service inspection was not performed on Nuclear Class II Components because it was not required at that stage of DAEC construction when it would have been used. For these components, shop and in-plant examination records of components and welds will be used as a basis for comparison with in-service inspection data.

The first 10-year interval for inservice inspections in accordance with the ASME Boiler and Pressure Vessel Code, Section XI commenced on February 1, 1975. This interval was extended for 9 months because of a 9 month outage for replacement of recirculation system inlet nozzle safe-ends in 1978-79. Therefore, the first 10-year interval ended on October 31, 1985.

The second 10-year interval for inservice inspections commenced on November 1, 1985 and is scheduled to end on October 31, 1995. The second 10-year inspection program addresses the requirements of the ASME Code, Section XI, 1980 Edition with Addenda through Winter 1981, subject to the limitations and modifications as stated in 10 CFR 50.55a.

Visual inspections for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the NRC and is delineated by Section XI of the ASME Code. These studies show that it requires thousands of stress cycles at stresses beyond those expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results, only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is fast and reliable. Surface inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing or radiography shall be used where defects can occur in concealed surfaces. Section 5.2.4 of the Updated FSAR provides details of the inservice inspection program.

Starting with the Cycle 9/10 Refueling Outage, an augmented inspection program was implemented to address concerns relating to Intergranular Stress Corrosion Cracking (IGSCC) in reactor coolant piping made of austenitic stainless steel. The augmented inspection program conforms to the NRC staff's positions set forth in Generic Letter 88-01 and NUREG-0313, Revision 2 for inspection schedule, inspection methods and personnel, and inspection sample expansion.

The first 10-year interval for inservice testing of pumps and valves in accordance with the ASME Code, Section XI commenced on February 1, 1975 and ended on January 31, 1985. The second 10-year inservice testing interval commenced on February 1, 1985 and is scheduled to end on January 31, 1995. The second 10-year testing program addresses the requirements of the ASME Code, Section XI, 1980 Edition with Addenda through Winter 1981, subject to the limitations and modifications of 10 CFR 50.55a. Section 3.9.6 of the Updated FSAR describes the inservice testing program.

3.6.H & 4.6.H BASES:

Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or other severe transient, while accommodating normal thermal motion during system startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of damage to piping as a result of a seismic or other event initiating dynamic loads or, in the case of a frozen snubber, exceeding allowable stress limits during system thermal transients. It is therefore required that all snubbers (safety-related) required to protect the Primary Coolant System or any other safety system or component be OPERABLE during reactor operation.

The Technical Specifications (TS) impose surveillance requirements for both visual inspections and functional testing of all safety-related snubbers. A visual inspection is the observation of the condition of installed snubbers to identify those that are damaged, degraded, or inoperable as caused by physical means, leakage, corrosion or environmental exposure. The performance of visual examinations is a separate process that compliments the functional testing program and provides additional confidence in snubber OPERABILITY.

The previous TS specified a schedule for snubber visual inspections that were based on the number of inoperable snubbers found during the previous visual inspection. Because the previous schedule for snubber visual inspections was based only on the number of inoperable snubbers found during the previous inspection, a large number of inoperable snubbers found resulted in the visual inspection schedule being excessively restrictive. This not only resulted in spending a significant amount of resources but also subjected plant personnel to unnecessary radiological exposure.

To alleviate this situation, the NRC developed an alternate schedule for visual inspections and issued it under Generic Letter 90-09, dated December 11, 1990. This alternate method maintains the same confidence level as the previous schedule and generally allows the performance of visual examinations and corrective action during plant outages.

The alternate inspection schedule is based on the number of unacceptable snubbers found during the previous inspection in proportion to the sizes of the various snubber population or categories. A snubber is considered unacceptable if it fails the acceptance criteria of the visual inspection. This inspection interval is based on a fuel cycle and may be as long as two fuel cycles.

When the cause for rejection of a snubber during visual inspection is clearly established and remedied for that snubber, and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to the cause of rejection of the snubber, or are similarly located or exposed to the same environmental conditions such as humidity, temperature, radiation, and vibration.

To verify that a snubber can operate within specific performance limits, a functional test is performed that typically involves removing the snubber and testing it on a specifically designed test stand. Functional testing provides a 95 percent confidence level that 90 percent to 100 percent of the snubbers operate within the specified acceptance limits.

To further increase the assurance of snubber reliability, functional tests will be performed once per OPERATING CYCLE. These tests will include stroking of the snubbers to verify proper movement, restraining characteristics and drag force (if applicable). Ten percent (10%) of the total of each type of snubber represents an adequate sample for such tests. Observed failures on these samples require testing of additional units.

The representative sample selected for functional testing shall represent the various configurations, operating environments and range of sizes of snubbers. At least 25 percent of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle.

2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.).

3. Snubbers within 10 feet of the discharge from a safety relief valve.

The 25 percent representative sample consists of those snubbers that meet the three categories above and have not been part of the last 3 representative samples.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Both the spare snubber and repaired/reinstalled snubber shall not be included in the sample plan. Failure of this functional test shall not be sole cause for increasing the sample size under the sample plan.

When a snubber is found inoperable, within 72 hours the subject snubber(s) are to be replaced or restored to OPERABLE status and an engineering evaluation performed. This evaluation is to determine the snubber mode of failure and identify any safety-related component or system that may have been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure adversely affected the supported component or system.

The TS action statements establish allowable outage times for systems or components addressed by the LCO. These time limits are applicable when the system or component is required to be OPERABLE and must be removed from service to perform required surveillance tests or repair/replacement of snubbers as discussed in TS SR 4.6.H.8. For snubbers, the allowable outage time is 72 hours. The 72 hour "clock" starts when the snubber is declared inoperable or when physical removal of the snubber has commenced. The 72 hour LCO is snubber specific. If snubber "A" is removed from service, its LCO time is 72 hours. If snubber "B" is removed from service, its 72 hour clock is independent from snubber "A". If a group of snubbers are removed simultaneously and replaced as a group, they need to be declared OPERABLE within the 72 hour limit.

In the event that the plant experiences a "potentially damaging transient," an inspection of the affected snubbers shall be performed. A "potentially damaging transient" is considered to be any event that causes physical damage to piping or component(s) that the snubber is supporting. The inspection requirements are specifically stated in TS Section 4.6.H.3.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of age and operating conditions. Due to implementation of the snubber service life monitoring program after several years of plant operation, the historical records to date may be incomplete.

The records will be developed from engineering data available. If actual installation data is not available, the service life will be assumed to commence with the initial criticality of the plant. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3.6 and 4.6 References

1) General Electric Company, Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for the Duane Arnold Energy Center, NEDE-30021-P, January, 1983.

2) "General Electric Boiling Water Reactor Increased Safety/Relief Valve Simmer Margin Analysis for Duane Arnold Energy Center," NEDC-30606, May, 1984.

3) General Electric Company, Duane Arnold Energy Center Reactor Pressure Vessel Fracture Toughness Analysis to 10 CFR 50, Appendix G, May 1983, NEDC-30839, December, 1984.

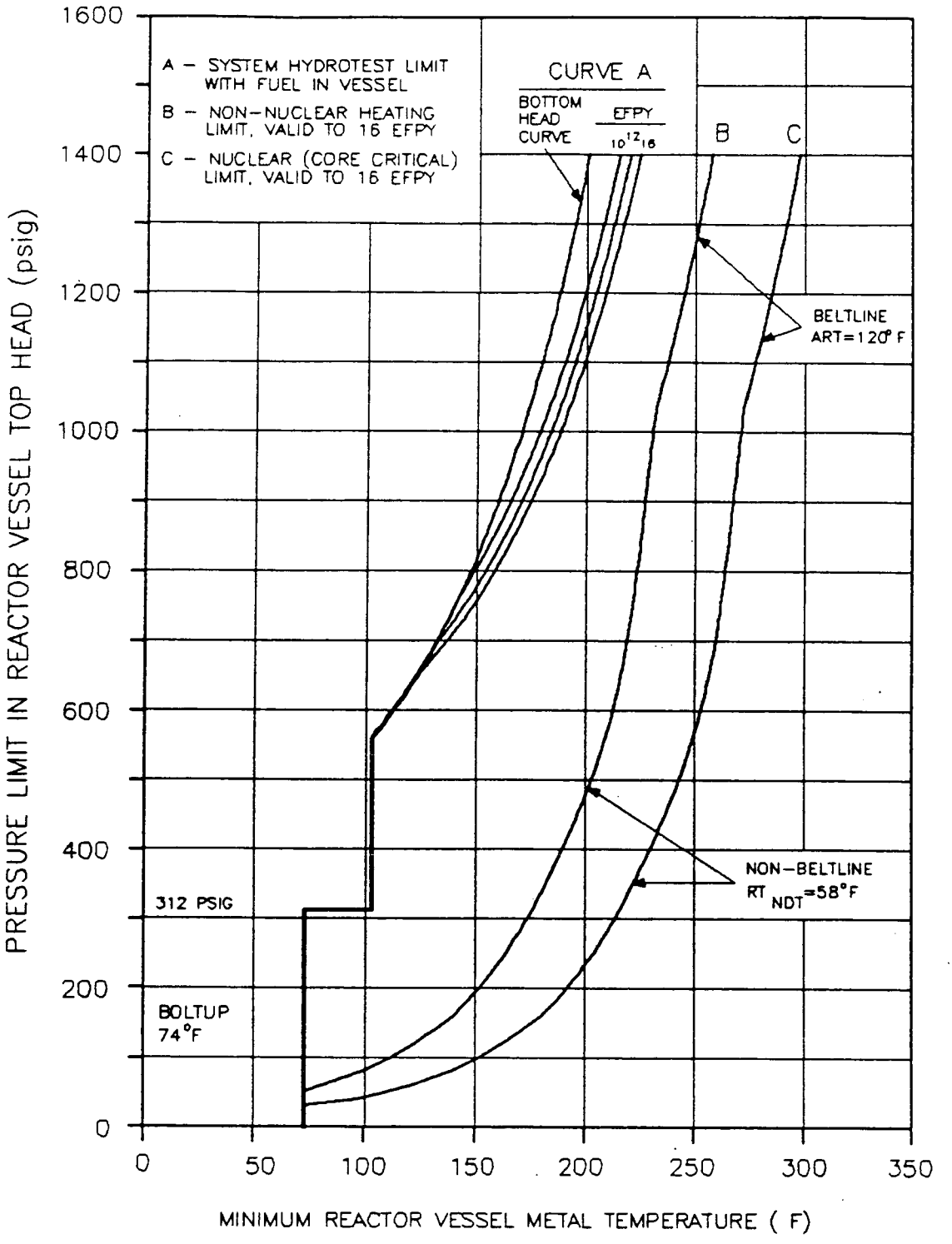

Figure 3.6-1: Pressure Versus Minimum Temperature Valid to Sixteen Full Power Years, per Appendix G of 10CFR50

SAFETY ASSESSMENT

1. INTRODUCTION

By letter dated June 4, 1993, Iowa Electric Light and Power Company requested changes to the Duane Arnold Energy Center (DAEC) Technical Specifications (TS) Section 3.6, "Primary System Boundary." These changes included: a) adding Limiting Conditions For Operation (LCOs), Surveillance Requirements (SRs), and shutdown requirements which incorporate requirements already specified in DAEC procedures, other DAEC TS Sections, and provide consistency with BWR Standard TS (NUREG-0123), and b) making administrative and minor editorial changes including reorganization, renumbering, denoting of defined terms, etc. to clarify and provide consistency with the other DAEC TS Sections, recently docketed DAEC TS submittals and BWR Standard TS.

2. ASSESSMENT

An independent evaluation of the DAEC TS was conducted as part of the DAEC TS Improvement Program. As a result of this evaluation, several TS Sections needed to be rewritten. TS Section 3.6 is one of these sections. The purpose of this rewrite was to improve clarity and consistency for the LCOs and SRs, provide consistent shutdown requirements, and make appropriate editorial changes. Improving the LCOs and SRs incorporates requirements already specified in the DAEC procedures, other TS sections, and the BWR Standard TS.

In addition, guidelines provided by NRC Generic Letters 90-09, "Alternate Requirements for Snubber Visual Inspection Intervals and Corrective Actions" and 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications" were incorporated. Incorporation of the Generic Letters enhances the existing TS by providing specific guidance for determination of the snubber inspection schedule based on snubber failures, and by removing redundant requirements for reactor vessel specimen withdrawal schedules.

Section 3.6.B, Coolant Chemistry, has been reformatted and revised placing many of the SR limits in tabular format. In addition, the LCOs and SRs have been revised providing consistent wording, specifying MODES of operation, and incorporating guidance provided by the Standard TS.

Based on the above assessment, we conclude that this request is acceptable.

ENVIRONMENTAL CONSIDERATIONS

10 CFR 5I.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluent that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. Iowa Electric Light and Power has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in Attachment 1, the proposed amendment does not involve a significant hazards consideration.

2. The proposed revisions to the limiting conditions for operation and surveillance requirements for Thermal and Pressurization Limitations, Coolant Chemistry, Coolant Leakage, Safety and Relief Valves, Jet Pumps, Jet Pump Flow Mismatch, Structural Integrity, and Shock Suppressors as discussed in DAEC TS Section 3.6, "Primary System Boundary" have no effect on the types or amounts of effluent released offsite.

3. The proposed revisions to the limiting conditions for operation and surveillance requirements for Thermal and Pressurization Limitations, Coolant Chemistry, Coolant Leakage, Safety and Relief Valves, Jet Pumps, Jet Pump Flow Mismatch, Structural Integrity, and Shock Suppressors as discussed in DAEC TS Section 3.6, "Primary System Boundary" have no effect on individual or cumulative occupational radiation exposure.