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Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
IES Utilities Inc.
Post Office Box 351
Cedar Rapids, IA 52406

SUBJECT: AMENDMENT NO. 203 TO FACILITY OPERATING LICENSE NO. DPR-49 - DUANE ARNOLD ENERGY CENTER (TAC NO. M85339)

Dear Mr. Liu:

The Commission has issued the enclosed Amendment No. 203 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications in response to your applications dated June 4, 1993, as supplemented February 14, 1994, and May 6, 1994.

The amendment revises the Technical Specifications (TS) by revising TS Section 3.6, "Primary Systems Boundary," adding definitions into Section 1.0, "Definitions," and revising the Bases to Section 3/4.6. This change will provide clarity and consistency of Limiting Conditions for Operations and Surveillance Requirements and become more consistent with Standard Technical Specifications. Some editorial and administrative changes were also made.

A copy of the related Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by Glenn B. Kelly
Glenn B. Kelly, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-331

Encls: 1. Amendment No. 203 to License No. DPR-49
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\DUANEARN\DUA85339.AMD
*See previous concurrence
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DATE	11/16/94		10/7/94		10/17/94		10/20/94		10/20/94	

OFFICE	*D:EMEB		*D:EMCB		D:PDIII-3		*OGC-OWFN	E	PM:PDIII-3	E
NAME	RWessman		JStrosnider		LNorrholm		EHoller		GKelly	
DATE	10/21/94		10/27/94		11/16/94		11/10/94		11/14/94	

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Mr. Lee Liu
IES Utilities Inc.

Duane Arnold Energy Center

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

IES UTILITIES INC.
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 203
License No. DPR-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by IES Utilities Inc., et al., dated June 4, 1993, as supplemented February 14, 1994, and May 6, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 203, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Leif J. Norrholm, Director
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: November 17, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 203

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove	Insert
ii	ii
vi	vi
1.0-10	1.0-10
3.6-1 through 3.6-3	3.6-1 through 3.6-3
3.6-3a through 3.6-3b	---
3.6-4 through 3.6-6	3.6-4 through 3.6-6
3.6-6a	---
3.6-7 through 3.6-13	3.6-7 through 3.6-13
3.6-13a	---
3.6-14 through 3.6-32	3.6-14 through 3.6-32
3.6-33 through 3.6-41	---

TECHNICAL SPECIFICATIONS
LIST OF TABLES (Continued)

40. SHUTDOWN MARGIN

SHUTDOWN MARGIN is the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are inserted, except for the analytically strongest worth control rod, which is fully withdrawn, with the core in its most reactive state during the OPERATING CYCLE.

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."

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 10 CFR 50, 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

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.

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.

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.

a. In RUN MODE:

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. Coolant Chemistry

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. Whenever the DOSE EQUIVALENT I-131 exceeds 0.6 μCi/ml, notify the USNRC as specified by 6.11.1.h.

2. The reactor coolant shall be determined to be within the specified chemistry limits by:

a. Measurement prior to pressurizing the reactor during each startup, if not performed within the previous 72 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

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.

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.

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.

SURVEILLANCE REQUIREMENTS

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.

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.

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

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.

2. With the conditions in Specifications 3.6.C.1.a, b, or c above not met, 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.

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.

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.

SURVEILLANCE REQUIREMENTS

C. Coolant Leakage

1. Reactor Coolant System leakage shall be checked by the Sump System and recorded at least once every 8 hours.

2. Verify Sump System OPERABILITY as specified in Table 4.2-E.

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.

LIMITING CONDITIONS FOR OPERATION

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.

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.

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.

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

SURVEILLANCE REQUIREMENTS

D. Safety and Relief Valves

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. At least one of the relief valves shall be disassembled and inspected once per OPERATING CYCLE.

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.

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

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.

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.

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.

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.

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.

SURVEILLANCE REQUIREMENTS

F. Jet Pump Flow Mismatch

1. Recirculation pump speed mismatch shall be verified at least once per day.

2. See Surveillance Requirement 4.3.F.4 for SLO requirements.

G. Structural Integrity

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. 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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

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.

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.

H. Shock Suppressors (Snubbers)

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.

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. 203.

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.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

induced snubber movement; (2) 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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

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.

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.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

c. Snubber release rate, where required, 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.

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. 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 and tested in accordance with 10CFR50, Appendix H. 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 provided 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.

Visual examinations for leaks will be made periodically on ASME Section XI Class 1, 2 and 3 systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the ASME Section XI boundaries.

The type of examination planned for each component depends on location, accessibility, and type of potential defect. Direct visual examination is proposed wherever possible since it is fast and reliable. Surface examinations are planned where practical, and where added sensitivity is required. Ultrasonic examination 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.

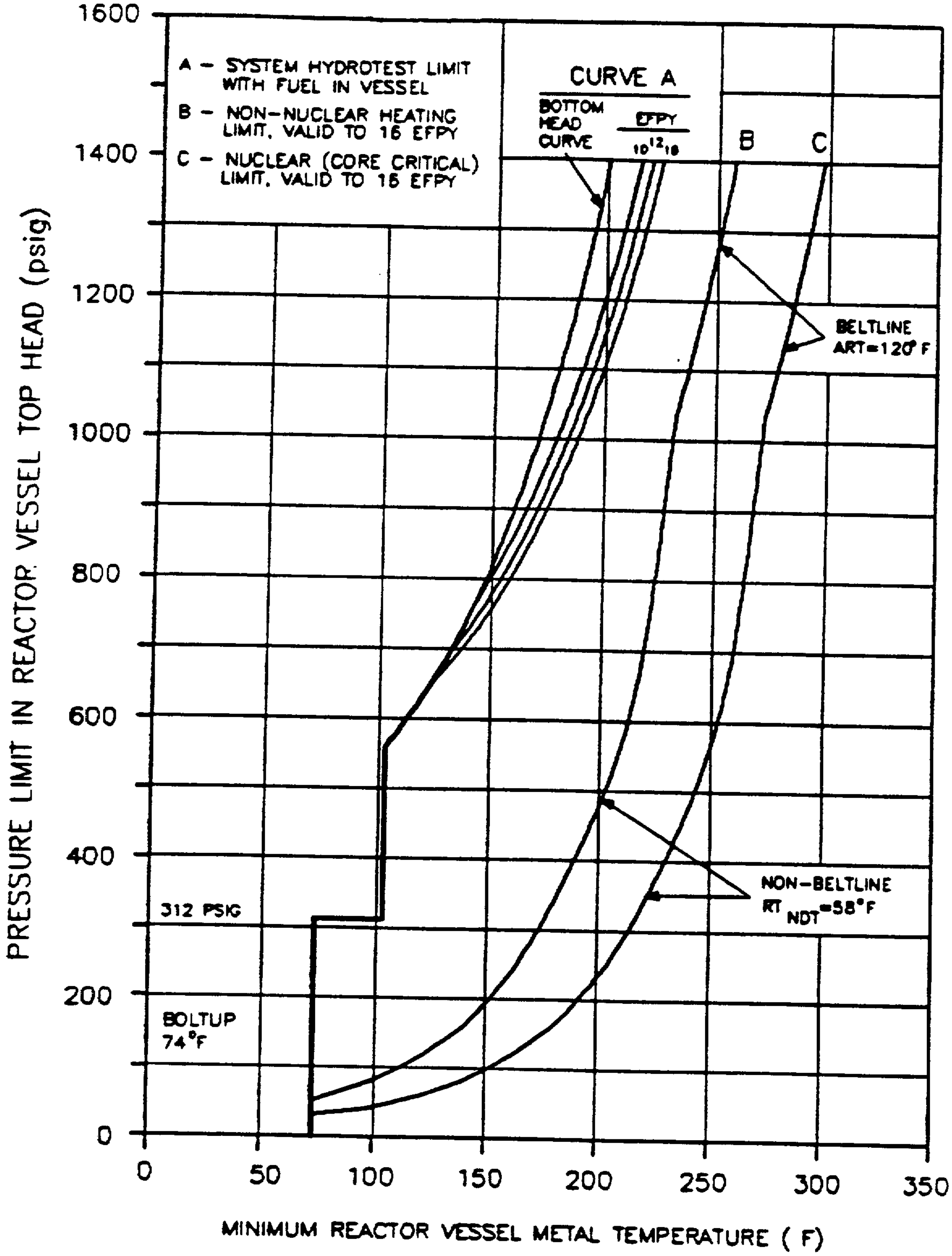
The records will be developed from engineering data available. If actual installation data is not available, the service life will be assumed to commence with the initial criticality of the plant. These records will provide statistical bases for future consideration of snubber service life. The requirements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



Pressure Versus Minimum Temperature Valid to Sixteen Full
Power Years, per Appendix G of 10 CFR 50

DUANE ARNOLD ENERGY CENTER IES UTILITIES INC. TECHNICAL SPECIFICATIONS
DAEC OPERATING LIMITS FIGURE 3.6-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 203 TO FACILITY OPERATING LICENSE NO. DPR-49

IES UTILITIES INC.

CENTRAL IOWA POWER COOPERATIVE

CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

In 1991, IES Utilities Inc. (the licensee), formerly known as Iowa Electric Light and Power Company, performed its own independent review of the Duane Arnold Energy Center (DAEC) Technical Specifications (TS) as part of a self-initiated TS improvement program. A portion of the program included comparison of the Duane Arnold TS with TS from similar plants, the General Electric Standard TS (NUREG-1202, July 1986), and the draft improved Standard TS (NUREG-1433). Based on that comparison, the licensee, by letter dated December 31, 1992, proposed changes to TS Section 3.6, "Primary System Boundary." Subsequent to that submittal, the licensee discovered some erroneous references and inconsistencies in their proposed changes. By letter dated June 4, 1993, the licensee corrected the errors and inconsistencies in the original submittal. The June 4, 1993, submittal, superseded the original submittal and this evaluation is based solely on the June 1993 submittal and TS Bases changes in the May 6, 1994, submittal. The submittal dated May 6, 1994, repeated a TS change (deletion of the inservice inspection interval start date) which was in the June 4, 1993, application, and provided 3/4.6 Bases changes.

The request for amendment proposes several changes to the DAEC Technical Specification (TS) Section 1.0, "DEFINITIONS," and a number of changes to Limiting Conditions of Operation (LCO) and Surveillance Requirements (SR) in TS Section 3/4.6, "PRIMARY SYSTEM BOUNDARY." These changes include appropriate revisions of the corresponding TS Bases Sections. The TS 3/4.6 LCO and SR Subsections that are affected are listed below:

- TS 3/4.6.A. - Thermal and Pressurization Limitations
- TS 3/4.6.B. - Coolant Chemistry
- TS 3/4.6.C. - Coolant Leakage
- TS 3/4.6.D. - Safety and Relief Valves
- TS 3/4.6.E. - Jet Pumps
- TS 3/4.6.F. - Jet Pump Flow Mismatch
- TS 3/4.6.G. - Structural Integrity
- TS 3/4.6.H. - Shock Suppressors

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The proposed changes to Section 3/4.6 are intended to clarify existing Limiting Conditions for Operation (LCO), Surveillance Requirements (SRs), add specific shutdown requirements and provide consistency with the rest of the plant TS and Standard TS. The proposed changes are plant specific in nature in that they clarify the existing TS and provide consistency in the overall Duane Arnold TS.

This amendment request also involves a substantial number of simple editorial changes to the TS 3/4.6. These changes include renumbering, capitalization of defined terms and replacing words to be consistent with the Standard TS. The existing wording is sometimes vague and misleading and could lead to misinterpretation. Index page v also corrected the page location of Figure 1.0-1. The staff, therefore, concludes that these editorial changes are acceptable throughout TS Section 3/4.6.C, as they are more specific and clearer than the existing TS and they would not affect the operation of the DAEC facility or adversely impact safety. These editorial changes will not be listed in this SER since they do not change the operation of the plant.

2.0 EVALUATION

2.1 TS Section 1.0 - DEFINITIONS

The DAEC Amendment request submittal proposes to add definitions for IDENTIFIED LEAKAGE, TOTAL LEAKAGE, UNIDENTIFIED LEAKAGE, AND DOSE EQUIVALENT I-131 to Section 1.0 of the DAEC TS, as proposed below:

- Definition 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 leakage detection systems."

- Definition 42, TOTAL LEAKAGE
  "TOTAL LEAKAGE is the sum of IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE."

- Definition 43, UNIDENTIFIED LEAKAGE
  "UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE."

- Definition 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 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 licensee proposes to add Definition 44, DOSE EQUIVALENT I-131, to provide clarity and consistency with LCO 3.6.B., "Coolant Chemistry," and the corresponding Standard TS definition. This definition is acceptable to the staff.

The licensee proposes to add Definition 41, IDENTIFIED LEAKAGE, Definition 42, TOTAL LEAKAGE and Definition 43, UNIDENTIFIED LEAKAGE, to provide clarity and consistency with LCO 3.6.C., "Coolant Leakage," and the corresponding Standard TS definition. Proposed Definitions 41, 42, and 43 are appropriate for LCO 3.6.C. as currently licensed, and for the proposed changes to LCO 3.6.C. The staff, therefore, finds the addition of definitions 41 through 44 acceptable.

2.2 TS 3/4.6.A. - Thermal and Pressurization Limitations

The licensee has proposed to revise TS Section 3.6.A.2 to delete from the LCO a commitment from the licensee to update TS Figure 3.6-1, Pressure versus Minimum Temperature Valid to Sixteen Full Power Years, per Appendix G of 10 CFR Part 50, prior to the expiration of 16 full power years, and to relocate the commitment to the Bases Section of the TS. This Figure gives the Pressure - Temperature Limit Curves for the DAEC reactor vessel. The staff approved Figure 3.6-1 on August 12, 1991. This commitment does not affect any surveillance requirements on the LCO, or provide any additional information to assist in the operation of the plant or in mitigating any accidents which could potentially occur. The staff has determined that this change is acceptable.

The licensee has proposed to revise TS Section 3.6.A.3 to identify the head flange and the shell adjacent to the head flange as the locations where temperature readings are to be taken before the licensee may place the reactor vessel head bolting studs in tension. These locations are acceptable to the staff and the change is acceptable.

The licensee has proposed to add Action Statement, TS Section 3.6.A.4, which requires the licensee to take action if any of the pressure - temperature (P/T) limits required by the TS Section 3.6.A.1., 2., or 3. are exceeded:

1. Average heatup/cooldown rate not to exceed 100 °F/hr

2. Reactor head required to be vented and no power operation allowed unless the reactor vessel temperature is greater or equal to Curve C in Figure 3.6-1

3. Reactor Vessel Studs shall not be placed in tension unless the temperature of the head flange and shell adjacent to the head flange are greater than 74 °F.

Should any of these limits be exceeded, the licensee is required to restore the temperature or pressure to the acceptable level within 30 minutes of discovery, and to perform an evaluation of the out-of-limit condition on the structural integrity of the Reactor Coolant System (RCS), in order to determine whether the RCS remains acceptable for continued operation. If these conditions cannot be met, the licensee will be required to be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours. The staff has reviewed the licensee's proposed Action Statement and has determined that it is more stringent than the comparable Action Statement in the Standard Technical Specifications. Therefore, this Action Statement is acceptable to the staff.

The licensee has proposed to apply the RUN, STARTUP, HOT SHUTDOWN, COLD SHUTDOWN, and REFUELING Modes of operation to TS Section 3.6.A.5, in regard to operation of the recirculation pump. These Modes of operation are consistent with the OPERATIONAL MODES in the Definitions Section of the TS and are acceptable to the staff.

The licensee has proposed to revise TS SR 4.6.A.2 to require that removal of vessel integrity specimens be done in accordance with 10 CFR Part 50 Appendix H. The SR also requires that the results of the specimen fluence determinations will be used to update TS Figure 3.6-1. This SR is in agreement with the requirements of Generic Letter 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications." This revision to TS SR 4.6.A.2, therefore, is acceptable to the staff.

2.3 TS 3/4.6.B. - Reactor Coolant Chemistry

The licensee proposed to revise the TS to clarify LCO Section 3.6.B.1 by revising the present LCO into three different sections.

Section 3.6.B.1.a will be revised to state:

- "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."

Section 3.6.B.1.b will be revised to state:

- "When in the RUN, STARTUP, or HOT SHUTDOWN MODE, 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 limit."

Section 3.6.B.1.c will be revised to state:

- "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."

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.

The surveillance requirements (SRs) for Section 4.6.B will be revised by placing the information in proposed tables or by formatting in accordance with the guidance provided by the standard TS.

SR 4.6.B.1.a will be revised to state: "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 will be 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 in the Standard TS.

SR 4.6.B.1.b, SR 4.6.B.1.c and SR 4.6.B.1.d will be deleted 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 in the Standard TS.

SR 4.6.B.1.e will be deleted and the sample requirements 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.

SR 4.6.B.1.f will be deleted and the sampling requirements 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.

SR 4.6.B.1.g will be deleted and the surveillance requirements and frequencies for sampling the Reactor Coolant System included in the proposed Table 4.6.B.1-1.

SR 4.6.B.1.h will be revised to SR 4.6.B.1.b and state: "Whenever the DOSE EQUIVALENT I-131 exceeds 0.6 μCi/ml, notify the USNRC as specified by 6.11.1.h." SR 4.6.B.1.e, referenced in the old SR 4.6.B.1.h will no longer exist; it will be replaced by information in Table 4.6.B.1-1. The words "(50% of the equilibrium value)" will also be deleted. Again, the Table provides adequate guidance as to the requirements. The unit of measurement "gm" will be 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. The word "by" replaces the word "in," as an editorial change.

The existing TSs were written when DAEC used gross iodine radiochemistry and sodium iodide detectors. Currently, DAEC is using the Dose Equivalent Methodology and no longer uses sodium iodide detectors. The Dose Equivalent Methodology provides a more quantitative and accurate analysis through the use of the isotopic analysis. The current Technical Specification calls for an isotopic analysis, as well as a gross measurement on each sample. The LCOs and SRs will be revised by placing the information in proposed Tables or by formatting in accordance with the guidance provided by the Standard TS. These revisions improve clarity, are consistent with current industry practices, and provide additional guidance not specifically stated in the existing DAEC Technical Specification. The proposed revision will also enhance the DAEC primary coolant chemistry surveillance program. The staff finds that the licensee's proposed Technical Specifications change, incorporating an improved isotopic analysis for dose equivalent Iodine-131 in the primary coolant system required by Technical Specification 3/4.6.B.1, is more conservative than the current Technical Specifications and is consistent with the Standard Technical Specifications for General Electric Boiling Water Reactors, NUREG-0123, Revision 3, and therefore acceptable.

The proposed changes move the existing LCO limits (LCO 3.6.B.2.a., b., and d., and LCO 3.6.B.3.a.) for conductivity, chloride content, and pH, applicable during operation, shutdown and refueling conditions, to TS Table 3.6.B.2-1. The limits proposed in Table 3.6.B.2-1 are equivalent or more conservative than the limits found in the existing LCOs 3.6.B.2.a., b., and d., and are acceptable to the staff. Under conditions when the unit is in the RUN Mode, the licensee is required to be in STARTUP Mode within 6 hours, should any of the applicable limits in Table 3.6.B.2-1 be exceeded for more than 72 continuous hours or more than 720 hours/year cumulatively. These changes are acceptable to the staff.

The proposed change also incorporates the shutdown requirements, during RUN or STARTUP MODE conditions, when chemistry or activity limits or surveillances are determined to be exceeded. The existing requirements on shutdown simply require the licensee to perform an orderly shutdown of the unit, without specifying the time in which shutdown is to be achieved. The proposed new shutdown action statements would require the licensee to be in HOT SHUTDOWN within 12 hours after determining that a chemistry or activity limit or surveillance was exceeded, and be in COLD SHUTDOWN within the next 24 hours, if the adverse situation was not corrected. The new shutdown requirements would be followed under the following conditions:

1. In the RUN Mode, if the conductivity, as measured at 25 °C, exceeds 10.0 μmhos/cm, or if the chloride content exceeds 500 ppb (i.e., maximum limits which require immediate shutdown if they are exceeded during RUN Mode conditions).

2. In STARTUP or HOT STANDBY Modes, if the applicable limits in Table 3.6.B.2-1 are exceeded.

The new chemistry shutdown requirements are more stringent than the comparable Action Statement in the NUREG-1202 (HOPE CREEK STANDARD TECHNICAL SPECIFICATIONS), and are therefore acceptable to the staff.

The proposed change also requires a water sample to be taken and reactor coolant chloride and pH analyses performed every 8 hours, whenever the conductivity exceeds the limits in Table 3.6.B.2-1. These changes are more stringent than previous requirements and are acceptable to the staff.

The proposed change revises the requirements for continuous conductivity monitoring found in existing LCO 3.6.B.3.b, and SRs 4.6.B.3.a. & 4.6.B.3.b., and has moved them into LCO 3.6.B.2.a.3 and SRs 4.6.B.2.e. and 4.6.B.2.f. The revisions make the following changes to the continuous conductivity requirements:

1. Reference to the locations of the continuous conductivity monitors would be moved to the TS Bases Section.

2. Channel checks of the continuous conductivity would be required once every 7 days.

3. In-line conductivity flow cells would be required to be installed and conductivity measurements taken when all three continuous conductivity monitors are determined to be inoperable, as opposed to when just one continuous conductivity monitor was inoperable.

The staff informed the licensee, as conveyed during the staff's conference call to IES on March 9, 1994, that the change in the requirements for installation of in-line flow cells appears to be in a non-conservative direction. The licensee forwarded information on March 14, 1994, to address the staff's question in regard to this matter. The licensee's information indicates that the actions required by the Plant Chemistry Procedure 2.13, Surveillance Test Procedure (STP) 46B003, and Chemistry Form 103 will be sufficient to account for the change in the requirement. The licensee will still be required by TS to maintain continuous conductivity monitoring, or else by the TS and the licensee's Chemistry Program to take appropriate corrective measures and actions when such monitoring is unavailable. In addition, moving the reference of the continuous conductivity monitor locations to the Bases Section does not affect operation or safety of the facility. Therefore, the staff concludes that the change in the requirement will not result in a significant reduction in the margin of safety for the plant, significantly increase the probability of an accident or malfunction of equipment important to safety, or create a new or different kind of accident from any accident previously evaluated. This change is, therefore, acceptable to the staff.

2.4 <u>TS 3/4.6.C. - Coolant Leakage</u>

The proposed change revises the applicability portion of LCO 3.6.C.1., "Coolant Leakage," from "Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212 °F," to "When in RUN, STARTUP or HOT SHUTDOWN MODE." These changes are consistent with the MODES of OPERATION in the Definitions Section of the TS and are acceptable to the staff.

The existing TS LCO 3.6.C.2 (renumbered 3.6.C.3) provides neither a reference nor specific requirements that define sump system operability. The proposed TS 3.6.C.3 references the applicable section of the TS Table 3.2-E which defines sump system operability. This proposed change also adds to the clarity of the TS and is, therefore, acceptable.

The licensee is currently required by existing LCO 3.6.C.3 to commence an orderly shutdown of the reactor to cold shutdown within 24 hours of determining that any of the leakage limits have been exceeded. In this amendment request, the licensee's June 4, 1993, application proposed revising the Action Statement as follows:

1. "With the conditions in Specifications 3.6.C.1.a, b, or c 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 COLD SHUTDOWN within the following 24 hours."

The licensee proposes making the change to the Action Statement on the basis that the change would provide clarity and consistency with the Standard TS. One of the original, proposed changes to the Action Statement was the addition of a 4-hour period which would allow the licensee to attempt bringing the leakage within proper limits, prior to commencing a shutdown of the reactor. Members of the staff informed the licensee during a conference call on February 7, 1994, that the additional 4 hours represented a reduction in the requirements. The staff based its assessment of the additional 4 hours on the following points:

1. The majority of licensees are prohibited by their TS to operate any unit with an identified Reactor Coolant Pressure Boundary (RCPB) leak. The 0 gpm RCPB leakage limit is consistent with the corresponding Specifications in the Hope Creek Standard TS (NUREG-1202) and the BWR-5 Standard TS (NUREG-0123).

2. By default, RCPB leakage at DAEC falls under the scope of TS Definition 43, "Unidentified Leakage." Therefore, under the current licensing basis, the licensee is allowed to operate the unit with a RCPB coolant loss of up to 5 gpm, and still not be required to shut down the reactor. This is less stringent than the industry norm. Although 5 gpm is very small in comparison to the total core inventory, a leak of this sort would still place the reactor in an extremely slow loss-of-coolant transient.

3. The additional 4-hour grace period to bring the leakage in line is a reduction, since in the case of identified RCPB leakage, the licensee is already allowed by its licensing basis to operate at power even with a 5 gpm RCPB leak in effect. Furthermore, the additional 4 hours is not consistent with the Action Statement in NUREG-1202, which states that, "With any Pressure Boundary Leakage, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours."

The licensee informed the staff, during the conversation of February 7, 1994, that it would withdraw the request for the additional 4-hour grace period. The licensee confirmed this by sending a revised LCO 3.6.C.2 to the staff on

February 14, 1994. LCO 3.6.C.2. will now read, "With the conditions in Specifications 3.6.C.1.a, b, or c above not met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours." This revision to proposed LCO 3.6.C.2., as written in the previous sentence, is consistent with the corresponding Hope Creek Standard TS Action Statement and is acceptable to the staff.

TS LCOs 3.6.C.4 and 3.6.C.5 will be added to identify specific actions to take in the event that the sump system is inoperable or if the sump system and the air sampling system are both inoperable. In the event that the sump system is inoperable, the air sampling system is verified operable and the sump system must be restored to an operable status within the next 24 hours or be in hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours. With both the sump system and air sampling system inoperable, the licensee has 4 hours to restore one system to an operable status, before initiating a shutdown. Proposed TS LCO 3.6.C.4 is basically a rewrite of the existing LCO 3.6.C.3 with the required actions to be taken more specifically described for clarification and to be consistent with other TS LCO requirements. The proposed TS LCO 3.6.C.5 will be added to cover a condition not previously addressed in the TS, i.e., the inoperability of the air sampling system, when it is required to be operable. The existing TS cannot readily be interpreted because they are rather vague and do not adequately address actions to be taken under certain circumstances. Therefore, the staff concludes that the proposed changes are necessary for clarity and are also consistent with leakage detection system LCOs in the Standard TS and in the TS of other BWRs and are, therefore, acceptable.

2.5 TS 3/4.6.D - Safety and Relief Valves (SRVs)

The existing TS LCO 3.6.D.1 will be revised by adding a note to state that SRVs which perform an Automatic Depressurization System (ADS) function must also satisfy OPERABILITY requirements as specified in TS 3.5.F. Existing TS LCO 3.6.D.2.a and b. will be revised to clarify the LCO requirements in the event that the safety function of relief valve(s) become inoperable. Also, existing LCO TS 3.6.D.3 will be revised to clarify and state the shutdown requirements when TS LCO 3.6.D.1 or 3.6.D.2 is not complied with. These changes to TS Section 3.6.D. are acceptable.

The licensee proposes changes to SRs 4.6.D.1, 4.6.D.2, 4.6.D.3, and 4.6.D.4 for the safety and relief valves. The proposed SR 4.6.D.1 permits that all safety and relief valves be tested, reinstalled or replaced with pretested spares once every 40 months. The existing SR 4.6.D.1 requires that all valves be tested every two cycles. This proposed change to allow the 40-month period to replace the existing two cycle test period requirement is consistent with the Standard TS and is, therefore, acceptable to the staff. The proposed SR 4.6.D.1 also requires that replacement spare valves be tested within the previous 40 months. This meets the minimum requirements of the ASME Code regarding testing frequency and is, therefore, acceptable. The proposed requirement to test at least one safety valve and three relief valves each cycle remains unchanged, as does the table of safety valve setpoints. There were several other minor wording changes proposed to these SRs to improve clarity and consistency with the Standard TS. These changes are only editorial in nature and are, therefore, acceptable.

2.6 <u>TS 3/4.6.E - Jet Pumps</u>

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

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

A shutdown requirement has been proposed for TS LCO 3.6.E.1.a.2 to be consistent with the guidance provided by the Standard TS and to eliminate unnecessarily cycling the plant to the COLD SHUTDOWN condition as currently required in the DAEC TS.

The staff finds these changes to TS Section 3/4.6.E acceptable.

2.7 <u>TS 3/4.6.F - Jet Pump Flow Mismatch</u>

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

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

Existing TS LCO 3.6.F.2 will be revised and renumbered as LCO 3.6.F.3.b.

Existing SR 4.6.F.1 will be editorially revised to provide additional clarification and consistency replacing the words "checked and logged" with "verified."

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

The staff finds these changes in TS Section 3/4.6.F acceptable.

2.8 TS 3/4.6.G - Structural Integrity

The licensee will combine the surveillance requirements for Inservice Inspection of ASME Code Class 1, 2, and 3 components (as required by existing SR 4.6.G.1.) and Inservice Testing requirements of ASME Code Class 1, 2, and 3 pumps and valves (as required by existing SR 4.6.G.2.) into proposed SR 4.6.G.1. The licensee has also proposed that the reference of the 2nd Ten Year Inservice Inspection interval and the 2nd Ten Year Inservice Testing interval (as referenced in SRs 4.6.G.1.a. and 4.6.G.2.a.) be removed. These changes do not affect safety and are acceptable to the staff.

The licensee has proposed revising the existing LCO 3.6.G., "Structural Integrity," into LCO 3.6.G.1. The existing LCO requires the structural integrity of the pressure boundaries be maintained in accordance with levels required by the original acceptance standard throughout the life of the plant.

LCO 3.6.G.1. now requires the structural integrity of ASME Section XI Code Class 1, 2, and 3 components be maintained in accordance with Inservice Inspection (for Code Class 1, 2, and 3 components) and Inservice Testing requirements (for Code Class 1, 2, and 3 pumps and valves) of Section XI of the ASME Boiler and Pressure Vessel Code, as required to be performed by SR 4.6.G.1. The proposed LCO is more specific than the existing requirement and is acceptable to the staff.

The licensee has proposed adding LCO 3.6.G.2., which will require that, under circumstances when the structural integrity of a ASME Code Class 1 or 2 component(s) does not conform to the requirements of the ASME Code Section XI, the structural integrity of the ASME Code Class 1 or 2 component(s) be restored within acceptable limits, or else that the affected component(s) be isolated prior to increasing the Reactor Coolant System (RCS) temperature above 212 °F. The licensee's basis for the addition was that it would make the requirements of TS Section 3.6.G. consistent with the corresponding requirements found in the Hope Creek Standard TS (NUREG-1202).

The staff informed the licensee, on March 10, 1994, that the RCS temperature, referred to for ASME Code Class 1 components in the corresponding Hope Creek Standard TS Action Statement, was 50 °F + Nil Ductility Temperature (NDT, in °F), not 212 °F. The licensee informed the staff, by conference call on March 24, 1994, that the standard operational practice during STARTUP at DAEC is to heat up the reactor prior to reaching (i.e., at a temperature very close to) 212 °F, with subsequent closing of the reactor head vent. This practice is delineated in a plant specific operating procedure. This practice will keep the licensee to the right of Curve C in TS Figure 3.6-1 during startup, and thus prevent the licensee from pressurizing the reactor during startup above the Curve C limiting pressure (i.e., starting up in the safe operating regime). The staff have therefore, concluded that the 212 °F temperature reference in the Class 1 component Action Statement is acceptable.

The staff concludes that LCO 3.6.G.2., as applied to ASME Code Class 2 components, is consistent with the intent of the applicable Action Statement for ASME Code Class 2 components in NUREG-1202 (HOPE CREEK STANDARD TS).

The licensee has proposed adding LCO 3.6.G.3., which will require that, under circumstances when the structural integrity of a ASME Code Class 3 component(s) does not conform to the requirements of the ASME Code Section XI, the structural integrity of ASME Code Class 3 component(s) be restored within acceptable limits, or else that the affected component(s) be isolated from service. LCO 3.6.G.3. is consistent with the corresponding Action Statement for ASME Code Class 3 components in TS Section 3.6.G. of the Hope Creek Standard TS (NUREG-1202). Therefore, the addition of proposed LCO 3.6.G.3. is acceptable to the staff.

The licensee has also proposed adding LCO 3.6.G.4., which would require that, with the reactor in the RUN, STARTUP, or HOT SHUTDOWN Modes of operation, should the requirements of proposed LCO 3.6.G.2. or 3.6.G.3. not be met, that the licensee perform an engineering evaluation to determine the effects of the component(s) condition for continued operation and determine that the component(s) remain acceptable for continued operation, or else isolate the affected component(s) and follow the applicable system LCO. The licensee's basis for the addition was that it would make the requirements of TS Section 3.6.G. consistent with the corresponding requirements found in the Hope Creek Standard TS (NUREG-1202).

The staff informed the licensee, during a conference call on February 7, 1994, that proposed LCO 3.6.G.4. was not consistent with the corresponding "Structural Integrity" Section in the Hope Creek Standard TS, and that the proposed addition could potentially conflict with the LCO Action Statements in Section 3.6.C., "Coolant Leakage," of the DAEC TS. The licensee informed the staff, during the conversation of February 7, 1994, that it would withdraw the request for the additional section, LCO 3.6.G.4. The licensee confirmed this by sending a revised LCO 3.6.G.4 to the staff on February 14, 1994.

2.9 TS 3/4.6.H - Shock Suppressors

The licensee proposed changes to TS LCOs 3.6.H.1 and 3.6.H.2 to provide clarity by spelling-out specific operation Modes and to correct the specification section number referenced for consistency throughout the revised TS section. Based on our review, we find that these changes are editorial in nature and are, therefore, acceptable.

The licensee proposed changes to TS SR 4.6.H.1 by replacing the existing visual examination schedule with an alternate visual examination schedule provided in Appendix B to Generic Letter 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions," dated December 11, 1990. The existing TS schedule for snubber visual examination is based on the number of inoperable snubbers found during the previous visual examination irrespective of the total population of snubbers. The proposed visual examination interval is based on the number of inoperable snubbers found during the previous visual examination in proportion to the size of the snubber population for each type of category. The purpose of the alternate inspection schedule is to allow the licensee to perform visual examination and corrective actions without reducing confidence level provided by the existing visual examination schedule. The existing inspection interval is based on a fuel cycle of 18 months. The proposed inspection interval is based on a fuel cycle of up to 24 months, and may be extended to as long as 48 months, depending on the number of unacceptable snubbers found during the previous visual examination. This change is consistent with Generic Letter 90-09 and is, therefore, acceptable to the staff.

The licensee proposed that TS SR 4.6.H.2 regarding visual inspection acceptance criteria be revised to include a requirement that a review and evaluation be performed and documented to justify continued operation with an unacceptable snubber. This proposed change is consistent with the requirement of the American Society of Mechanical Engineers (ASME) Operation and Maintenance Code (OM Code) regarding supported systems or components of which an unacceptable snubber is a part, and is, therefore, acceptable.

The licensee proposed to add SR 4.6.H.3, "Transient Event Inspection." Existing SRs 4.6.H.3, 4.6.H.4 and 4.6.H.5 will be renumbered to SRs 4.6.H.4, 4.6.H.5 and 4.6.H.6, respectively. The proposed SR 4.6.H.3 requires that in the event of a potential damaging transient, an inspection of all affected snubbers be performed and freedom-of-motion of affected mechanical snubbers be verified within 72 hours for accessible systems and 6 months for inaccessible systems following such an event. This proposed change is consistent with the Standard TS and with the ASME OM Code requirement regarding transient dynamic events (e.g., water hammer, steam hammers) that may affect snubber operability and is, therefore, acceptable. The other changes in the sections mentioned above are editorial in nature and are, therefore, acceptable.

The licensee proposed to add SRs 4.6.H.7 and 4.6.H.8 and to delete the existing SR 4.6.H.6 for the snubber service life monitoring. The proposed SR 4.6.H.7 requires that replacement and repaired snubbers be tested to meet the functional test criteria before installation in the unit. This proposed change is consistent with the ASME OM Code snubber inservice inspection requirements regarding the replacement and modification of snubbers and is, therefore, acceptable. With regard to the service life monitoring, both the existing SR 4.6.H.6 and the proposed SR 4.6.H.8 require that the service life of all snubbers be monitored, and that documentation be maintained to provide assurance that the service life of each snubber is not exceeded between surveillance inspection. In addition, the existing SR 4.6.H.6 requires that in the event that the indicated service life will be exceeded, prior to the next scheduled inspection, the snubber be reevaluated, replaced or reconditioned to extend its service life beyond the next scheduled inspection. The proposed SR 4.6.H.8 requires that all critical parts of the snubber be monitored and replaced as necessary, so that the snubber service life will not be exceeded during a period when the snubber is required to be operable. This proposed SR 4.6.H.8 appears to have incorporated the intent of the existing SR 4.6.H.6 regarding the snubber service life monitoring and is, therefore, acceptable.

2.10 Bases Sections 3/4.6.A-H

The Bases Sections for 3/4.6.A-E, G & H have been revised to reflect the proposed changes and are acceptable. Bases Section 3/4.6.F was not changed.

2.11 Supplemental Information

The supplemental information provided on February 14, 1994, did not change the proposed no significant hazards determination.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Iowa State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (58 FR 39052). Accordingly, the 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.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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