PROPOSED CHANGE RTS 129 TO

DUANE ARNOLD ENERGY CENTER

TECHNICAL SPECIFICATIONS

The holders of License DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with the attached new proposed pages. A list of the affected pages is included.

The justification for this change is to delete material which is no longer relevant, clarify reporting times and bring submittal dates for filing reports into compliance with the Code of Federal Regulations. Additionally, this proposed change will correct an administrative error in Table 3.7-3.

The following is a summary of the proposed changes:

Table 6-11-1 will be revised in order to bring it into compliance with the Code of Federal Regulations, to clarify reporting times and to delete information which is no longer relevant. These changes include:

* Revision of the reporting dates for 10CFR40.64(b) and 10CFR70.53 to reflect the reporting dates defined in the Code of Federal Regulations.

* Specify the reporting time for Changes, Tests, and Experiments as being within 60 days after January 1.

* Delete the reference to the Startup Report and First Year Operation Report due to the fact that they are no longer relevant.

Section 6.12, Initial Criticality, will be deleted as being no longer relevant.

Table 6.11-2 will be revised to reflect the reports and timing for submittal of written reports as indicated in section 6.11.2. This includes changing the Reportable Occurrence submittal time from 10 days to 14 days and changing the Unusual Event Report to a Reportable Occurrence with a 30 day written report requirement.

Table 3.7-3 will be revised to add Drywell Equipment Drain Discharge to the list of Primary Containment Power Operated Isolation Valves. These two valves, in isolation group 2 were inadvertently omitted from RTS-125 which was approved as Amendment 66.

LIST OF AFFECTED PAGES

6.12-1
6.11-12
6.11-13
6.11-14
3.7-25

TABLE 6-11-1

REPORTING SUMMARY - ROUTINE REPORTS

Requirement	Report	Timing of Submittal
TS	Annual Exposure	Within 60 days after January 1.
§20.407	Personnel Exposure and Monitoring	Within first quarter of each calendar year.
§20.408	Personnel Exposure on Termination of Employment or Work	Within 30 days after the exposure of the individual has been determined or 90 days after date of termination of employment or work assignment, whichever is earlier.
§40.64(a)	Transfer of Source Material	Promptly upon transfer.
§40.64(a)	Receipt of Source Material	Within 10 days after material is received.
§40.64(b)	Source Material Inventory	Within 30 days after September 30 of each year.

TABLE 6-11-1 (cont)

REPORTING SUMMARY - ROUTINE REPORTS

Requirement	Report	Timing of Submittal
§50.59(b)	Changes, Tests, and Experiments	Within 60 days after January 1.
§70.53	Special Nuclear Material Status	Within 30 days after March 31 and September 30 of each year.
§70.54	Transfer of Special Nuclear Material	Promptly upon transfer
§70.54	Receipt of Special Nuclear Material	Within 10 days after material is received
Appendix G to 10 CFR Part 50	Fracture Toughness	On an individual-case basis at least 3 years prior to the date when the predicted fracture toughness levels will no longer satisfy section V.B. of Appendix G to 10 CFR Part 50.
Appendix H to 10 CFR Part 50	Reactor Vessel Material Surveillance	Completion of tests after each capsule withdrawal.
Appendix J to 10 CFR Part 50	Reactor Containment Building Integrated Leak Rate Test	Approximately 3 months following conduct of test.

[1] Technical Specifications

TABLE 6.11-2

REPORTING SUMMARY - NONROUTINE REPORTS

Requirement	Report	Notification	Initial Written Report Within 14 days	15 days	30 days	3 mo
TS[1]	Reportable Occurrence	Within 24 hours	X			
TS	Reportable Occurrence				X	
§20.405	Overexposures and Excessive Levels of Radiation and Concentration of Radioactive Material				X	
§20.402	Theft or loss of Material	Immediately			X	
§20.403(a)	Severe Accident Involving Licensed Material	Immediately				
§20.403(b)	Accident Involving Licensed Material	Within 24 hours				
§40.64(c)	Theft or Unlawful Diversion of Source Material	Promptly		X		
§50.59(d)	Authorization of Changes, Tests, and Experiments	X[2]				
§70.52	Accidental Criticality or Loss of Special Nuclear Material	Promptly				
§73.42	Unaccounted for Shipments, Suspected Theft, or Unlawful Diversion of Special Nuclear Material	Immediately	X			
TS	Unique					X[3]

TABLE 3.7-3

PRIMARY CONTAINMENT POWER OPERATED ISOLATION VALVES

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves	Maximum Operating Time (Second)	Normal Position	Action on Initiating Signal
1	*Main Steam Line	8	3<T<5	O	GC
1	Main Steam Line Drain	2	15	C	SC
1	Recirculation Loop Sample	2	NA	C	SC
3	Recirculation Pump Seal Purge	2	5	O	GC
3	O_2 Analyzer	20	NA	O	GC
2	Drywell Floor Drain Discharge	2	4	O	GC
2	Drywell Equipment Drain Discharge	2	4	O	GC
3	Drywell Purge Inlet	1	5	C	SC
3	Drywell Purge Outlet	3	5	C	SC
3	Torus Purge Outlet	3	5	C	SC
3	Drywell and Torus Nitrogen Makeup	2	NA	O	GC
4	RHR Shutdown Cooling Supply	2	22	C	SC
3	*Containment Compressor Suction	2	25	O	GC
3	Suppression Pool/Drywell and Suppression Pool Purge Inlet	2	5	O	GC

DAEC-1

PROPOSED CHANGE RTS 128 TO

DUANE ARNOLD ENERGY CENTER

TECHNICAL SPECIFICATIONS

The holders of License DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with the attached new proposed pages. A list of the affected pages is included.

The justification for this change is the NRC letter of November 20, 1980 which provided several changes, clarifications and improvements from the previous standard technical specifications as well as including surveillance requirements for mechanical snubbers.

The following is a summary of the proposed changes.

> Section 3.6.H and Tables 4.6-3 and 4.6-4 will be revised to incorporate surveillance requirements for mechanical snubbers. The proposed changes also provide clarification to the visual inspection and functional test acceptance criteria as well as snubber life monitoring requirements. This change includes changing the page numbers of several pages listed in the List of Affected Pages. Additionally, the BASES for section 3.6.H and 4.6.H will be revised to reflect changes to the subject technical specifications. In accordance with 10CFR50.36 (a), the technical specification bases shall not become part of the technical specifications.

LIST OF AFFECTED PAGES

REMOVE PAGES	INSERT PAGE
3.6-10	3.6-10
3.6-11	3.6-11
	3.6-12
	3.6-13
	3.6-14
	3.6-15
3.6-12	3.6-16*
3.6-13	3.6-17*
3.6-14	3.6-18*
3.6-15	3.6-19*
3.6-16	3.6-20*
3.6-17	3.6-21*
3.6-18	3.6-22*
3.6-19	3.6-23*
3.6-20	3.6-24*
3.6-21	3.6-25*
3.6-22	3.6-26*
3.6-23	3.6-27*
3.6-24	3.6-28*
3.6-25	3.6-29*
3.6-26	3.6-30*
3.6-27	3.6-31*
3.6-28	3.6-32*
3.6-28a	3.6-33*
3.6-29	3.6-34*
3.6-30	3.6-35*
3.6-31	3.6-36*
3.6-32	3.6-37
3.6-32a	3.6-38
3.6-32b	3.6-39
3.6-33**	3.6-40
3.6-34**	3.6-41
3.6-35**	
3.6-36**	
3.6-37**	
3.6-38**	
3.6-39**	
3.6-40**	
3.6-41	3.6-42
3.6-42	3.6-43
3.6-43	3.6-44
3.6-44	3.6-45
	3.6-46
	3.6-47
3.6-45	3.6-48*
Figure 3.6-1	3.6-49*

* Page Number Change Only
** Page Currently Blank

LIMITING CONDITIONS FOR OPERATION

H. Shock Suppressors (Snubbers)

1. During all modes of operation, except Cold Shutdown and Refuel, all safety related snubbers listed in Tables 4.6-3 and 4.6-4 shall be operable, except as noted in 3.6.H.2 through 3.6.H.4 below.

2. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeding 72 hours unless the snubber is sooner made operable or replaced.

3. If the requirements of 3.6.H.1 and 3.6.H.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

5. Snubbers may be added to safety related systems without prior License Amendment to Tables 4.6-3 or 4.6-4 provided that a revision to Table 4.6-3 or 4.6-4 is included with the next License Amendment request.

SURVEILLANCE REQUIREMENTS

H. Shock Suppressors (Snubbers)

Each snubber listed in Tables 4.6-3 and 4.6-4 shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

1. Visual Inspections

The inservice visual inspection of snubbers shall be implemented by completion of a visual inspection of all snubbers listed in Tables 4.6-3 and 4.6-4, and shall subsequently be performed in accordance with the following schedule:

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

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

All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	Snubbers are categorized in two groups, "accessible and inaccessible," based on their accessibility for inspection during reactor operation. These two groups will be inspected independently according to the above schedule.
	2. Visual Inspection Acceptance Criteria
	Visual inspection shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) (for hydraulic snubbers) inspection of the hydraulic fluid reservoir and fluid connections, (3) attachments to the foundation or supporting structure are secure, and (4) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movemet and is not frozen up. Snubbers which appear inoperable, as a result of visual inspection, may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per specifications 4.6.H.4 or 4.6.H.5 as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	3. Functional Tests At least once per 18 months during shutdown, a representative sample (10% of the total of each type of snubber in use in the plant) shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of specification 4.6.H.4 or 4.6.H.5, an additional 10% of that type of snubber shall be functionally tested. The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% 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 Snubbers identified in Table 4.6-5 as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample. 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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling. If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria. For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service. 4. Hydraulic Snubbers Functional Test Acceptance Criteria The hydraulic snubber functional test shall verify that: 1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	2. Snubber bleed, or release rate where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.
	Mechanical Snubbers Functional Test Acceptance Criteria
	5. The mechanical snubber functional test shall verify that:
	1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.
	2. Drag force on any snubber shall not have increased more than 50% since the last functional test. If drag force has increased more than 50% but is still less than the specified maximum drag force, the affected snubber shall be functionally tested at the next test interval (in addition to the normal representative sample), but shall not be considered as a failed snubber under specification 4.6.H.3.
	3. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

4. 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.

6. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber listed in Table 4.6-3 and 4.6-4 shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

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.

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.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 546°F. Calculated stresses were 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 analysis as shown in FSAR Appendix K. "Field Fabricated Reactor Vessel". This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

The permissible flange to adjacent shell temperature differential of 145°F is the maximum calculated for 100°F hour heating and cooling 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 heating 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 environs. 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 nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons (> 1 mev) above about 10^{17} nvt may shift the NDT temperature of the vessel base metal above the initial value. Extensive tests have established the magnitude of changes as a function of the integrated neutron exposure.

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 wires and samples will be

removed and tested according to 10 CFR 50 Appendix H. Results of these analyses will be used to adjust Figure 3.6-1 as appropriate.

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. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

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 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 indicate that the iodine

transients during power generation are less than a factor of ten. Iodine concentrations behavior during transients is under AEC review. 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 (.2-.3 ppm) 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 .2-.3 ppm, such as refueling, reactor startup and hot standby. During these periods, a limit of 0.1 ppm 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-standard condition include operation of the

reactor cleanup 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 cleanup system to re-establish the purity of the reactor coolant. During some periods of operation, conductivity or chloride concentration may exceed 5.0 µmho/cm or 0.2 ppm 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.

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 stack offgas 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 4 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.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pumps is 50 gpm and the capacity of the drywell equipment sump pumps is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with margin.

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.

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 design basis 4.4.4.1 of Subsection 4.4 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 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Report in FSAR Amendment No. 3 (response to AEC Question H.1.1) 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 analyses of the plant isolation transients are evaluated in each reload analyses. These analyses show 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 anticipated normal transient. These analyses verify that peak system pressure is limited to greater than a 125 psi margin to the allowed vessel overpressure of 1375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year 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.

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. Therefore, if a failure occurs, repairs must be made to assure the validity of the calculated consequences.

The following factors form the basis for the surveillance requirements:

a. A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

b. The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

c. The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Section 4.6.G.1 and 2.

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. Thus the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing 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.

3.6.F & 4.6.F BASES

Jet Pump Flow Mismatch

The LPCI loop selection logic has been previously described in the DAEC FSAR. 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 pump, the loop select logic trips that pump before making the loop selection.

An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, continuous operation under such conditions is not appropriate. The reactor may, however, be operated up to 24 hours with one recirculation loop out of service. This short period of time permits corrective action to be taken. During this period the reactor will be operated within the restrictions of the thermal analysis and will be protected from fuel damage resulting from anticipated transients.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of faster pump is below 50% of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G & 4.6.G BASES:

REACTOR COOLANT SYSTEM

Structural Integrity

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

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

The engineering and design effort associated with the Duane Arnold Energy Center predates the availability of the ASME Inspection Code. However, this Code, including subsequent Addendum through the Winter 1972 Addenda, dated December 31, 1972, has been used as a guide in the preparation of the DAEC In-Service Inspection Plan for Nuclear Class I and Class II Components for the first 40-month interval of the 10-year program, and maximum access has been provided to the extent drywell design and radiation levels permit.

Inspections and testing concluded subsequent to the first 40-month interval are as required by 10 CFR 50, Section 50.55a(g).

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

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is fast and reliable. Surface inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing or radiography shall be used where defects can occur in concealed surfaces. Appendix J of the DAEC FSAR provides details of the inspection program for the first 40-month cycle.

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 severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.H.4 prohibits startup with inoperable snubbers.

All safety related snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level, when applicable, and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests or analysis to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed.

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories (Reference 1) has shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are avaiable, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

To further increase the assurance of snubber reliability, functional tests will be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent of the total of each type of snubber represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units.

When the cause of the rejection of a snubber 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 rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

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 their 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.

Ref. 1 - Report H. R. Erickson, Bergen Paterson, to K. R. Goller, NRC, October 7, 1974. Subject: Hydraulic Shock Sway Arrestors.

TABLE 4.6-3

SNUBBERS ACCESSIBLE DURING NORMAL OPERATION

Identification No.	System	Location
GBC-1-SS-56	RHR Service Water	Reactor
GBC-1-SS-57	RHR Service Water	Reactor
GBC-2-SS-62	RHR Service Water	Reactor
HCC-8-SS-11	Core Spray Pump Suction	Reactor
HCC-8-SS-12	Core Spray Pump Suction	Reactor
EBB-16-SS-231	RHR	Reactor
EBB-16-SS-232(2 ea.)*	RHR	Reactor
EBB-16-SS-233	RHR	Reactor
EBB-16-SS-234(2 ea.)*	RHR	Reactor
GBB-3-SS-235	RHR	Reactor
GBB-3-SS-236	RHR	Reactor
GBB-3-SS-237	RHR	Reactor
GBB-3-SS-238	RHR	Reactor
GLE-8-SS-239	RHR	Reactor
GLE-8-SS-240	RHR	Reactor
GBB-10-SS-241	RHR	Reactor
GBB-10-SS-242(2 ea.)*	RHR	Reactor
GBB-10-SS-243	RHR	Reactor
GBB-4-SS-210	RHR	Reactor
GBB-4-SS-211	RHR	Reactor
GBB-4-SS-212	RHR	Reactor
GBB-4-SS-213	RHR	Reactor
GBB-16-SS-214	RHR	Reactor
GBB-5-SS-215	RHR	Reactor
GBB-4-SS-216(2 ea.)*	RHR	Reactor
GBB-4-SS-217(2 ea.)*	RHR	Reactor
HBB-21-SS-218(2 ea.)*	RHR	Reactor
HBB-23-SS-219	RHR	Reactor
HBB-23-SS-220	RHR	Reactor
HBB-24-SS-221	RHR	Reactor
HBB-24-SS-222	RHR	Reactor
GBB-7-SS-223	RHR	Reactor
GBB-7-SS-224	RHR	Reactor
GBB-6-SS-225	RHR	Reactor
GBB-6-SS-226	RHR	Reactor
HBB-24-SS-227(2 ea.)*	RHR	Reactor
HBB-24-SS-228(2 ea.)*	RHR	Reactor
HBB-24-SS-229	RHR	Reactor
HBB-29-SS-199	RHR	Reactor
HBB-30-SS-205	RHR	Reactor
HBB-30-SS-206	RHR	Reactor
HBB-30-SS-245	RHR	Reactor

*(2 ea.) - Indicates there are 2 snubbers with that number.

TABLE 4.6-3 (cont.)

Identification No.	System	Location
HBB-7-SS-17	RCIC Turbine Exhaust	Reactor
HBB-7-SS-18	RCIC Turbine Exhaust	Reactor
HBB-7-SS-19(2 ea.)*	RCIC Turbine Exhaust	Reactor
GBB-13-SS-16	Core Spray	Reactor
GBB-14-SS-20	Core Spray	Reactor
HBB-2-SS-7	Core Spray	Reactor
HBB-2-SS-8	Core Spray	Reactor
HBB-1-SS-9	Core Spray	Reactor
HBB-1-SS-10	Core Spray	Reactor
EBB-14-SS-13	HPCI Steam Supply	Reactor
EBB-14-SS-14	HPCI Steam Supply	Reactor
EBB-14-SS-15	HPCI Steam Supply	Reactor
EBB-14-SS-16	HPCI Steam Supply	Reactor
EBB-14-SS-16A	HPCI Steam Supply	Reactor
HBB-6-SS-20	HPCI Turbine Exhaust	Reactor
HBB-6-SS-22	HPCI Turbine Exhaust	Reactor
HBD-31-SS-71	Emergency Service Water	Reactor
HBD-31-SS-101	Emergency Service Water	Reactor
HBB-25-SS-178	Fuel Pool to RHR	Reactor
GBD-29-SS-12	Aux. Boiler Stm. to HPCI	Reactor
DCB-2-SS-78	RWCU	Reactor

*(2 ea.) - Indicates there are 2 snubbers with that number.

Modifications to this Table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

TABLE 4.6-4

SNUBBERS INACCESSIBLE DURING NORMAL OPERATION

Identification No.	System	Location
DLA-5-SS-10	RHR	Drywell
DLA-5-SS-11	RHR	Drywell
DLA-6-SS-12	RHR	Drywell
DLA-6-SS-13	RHR	Drywell
DLA-4-SS-14	RHR	Drywell
DLA-4-SS-15	RHR	Drywell
DBA-4-SS-35	RCIC	Drywell
DBA-4-SS-36	RCIC	Drywell
DLA-3-SS-1	HPCI Steam Supply	Drywell
DLA-3-SS-2	HPCI Steam Supply	Drywell
DLA-3-SS-3	HPCI Steam Supply	Drywell
DBA-6-SS-29	CRD	Drywell
DBA-6-SS-30	CRD	Drywell
DCA-6-SS-48	RWCU	Drywell
DCA-6-SS-49	RWCU	Drywell
DCA-6-SS-50	RWCU	Drywell
DBA-4-SS-34	RCIC Steam Supply	Drywell
DBA-5-SS-31	Head Spray	Drywell
DBA-5-SS-37	Head Spray	Drywell
DBA-5-SS-38	Head Spray	Drywell
DBA-5-SS-47	Head Spray	Drywell
DLA-2-SS-4	RHR	Drywell
DLA-2-SS-5	RHR	Drywell
DLA-2-SS-6	RHR	Drywell
DLA-2-SS-7	RHR	Drywell
DLA-2-SS-8	RHR	Drywell
DLA-2-SS-9	RHR	Drywell
DBA-7-SS-71	RCIC to FW Line	Steam Tunnel
DCA-14-SS-72	RWCU to FW Line	Steam Tunnel
GBC-6-SS-17	Main Steam Relief	Drywell
GBC-6-SS-250	Main Steam Relief	Drywell
GBC-6-SS-251	Main Steam Relief	Drywell
GBC-6-SS-252	Main Steam Relief	Drywell
GBC-6-SS-253	Main Steam Relief	Drywell
GBC-6-SS-254	Main Steam Relief	Drywell
GBC-7-SS-18	Main Steam Relief	Drywell
GBC-7-SS-19	Main Steam Relief	Drywell
GBC-8-SS-20	Main Steam Relief	Drywell
GBC-8-SS-21	Main Steam Relief	Drywell
GBC-8-SS-44	Main Steam Relief	Drywell

*(2 ea.) - Indicates there are 2 snubbers with that number.

TABLE 4.6-4 (cont)

Identification No.	System	Location
GBC-8-SS-45	Main Steam Relief	Drywell
GBC-8-SS-46	Main Steam Relief	Drywell
GBC-9-SS-22	Main Steam Relief	Drywell
GBC-9-SS-23	Main Steam Relief	Drywell
GBC-9-SS-41	Main Steam Relief	Drywell
GBC-9-SS-271	Main Steam Relief	Drywell
GBC-9-SS-272	Main Steam Relief	Drywell
GBC-9-Ss-273 (2 ea)*	Main Steam Relief	Drywell
GBC-9-SS-274	Main Steam Relief	Drywell
GBC-9-SS-275	Main Steam Relief	Drywell
GBC-10-SS-24	Main Steam Relief	Drywell
GBC-10-SS-25	Main Steam Relief	Drywell
GBC-10-SS-269	Main Steam Relief	Drywell
GBC-10-SS-270 (2 ea)*	Main Steam Relief	Drywell
GBC-10-SS-276	Main Steam Relief	Drywell
GBC-11-SS-26	Main Steam Relief	Drywell
GBC-11-SS-27	Main Steam Relief	Drywell
GBC-11-SS-32	Main Steam Relief	Drywell
GBC-11-SS-255	Main Steam Relief	Drywell
GBC-11-SS-256	Main Steam Relief	Drywell
GBC-11-SS-257	Main Steam Relief	Drywell
GBC-11-SS-258 (2 ea)*	Main Steam Relief	Drywell
SSB-1-MS	Main Steam	Drywell
SSB-2-MS	Main Steam	Drywell
SSC-1-MS	Main Steam	Drywell
SSC-2-MS	Main Steam	Drywell
SSA-1-MS	Main Steam	Drywell
SSA-2-MS	Main Steam	Drywell
SSD-1-MS	Main Steam	Drywell
SSD-2-MS	Main Steam	Drywell
SSA-1	Recirc	Drywell
SSB-1	Recirc	Drywell
SSA-2	Recirc	Drywell
SSB-2	Recirc	Drywell
SSA-3	Recirc	Drywell
SSB-3	Recirc	Drywell
SSA-4	Recirc	Drywell
SSB-4	Recirc	Drywell
SSA-5	Recirc	Drywell
SSB-5	Recirc	Drywell
SSA-6	Recirc	Drywell
SSB-6	Recirc	Drywell

*(2 ea.) - Indicates there are 2 snubbers with that number.

TABLE 4.6-4 (cont)

Identification No.	System	Location
SSA-7	Recirc	Drywell
SSB-7	Recirc	Drywell
SSA-8	Recirc	Drywell
SSB-8	Recirc	Drywell
SSA-9	Recirc	Drywell
SSB-9	Recirc	Drywell
SSA-10 (2 ea)*	Recirc	Drywell
SSB-10 (2 ea)*	Recirc	Drywell
SSA-11	Recirc	Drywell
SSB-11	Recirc	Drywell
DBA-3-SS-1	MSIV Leakage Control Line	Drywell
DBA-4-SS-1	MSIV Leakage Control Line	Drywell
DBA-4-SS-2	MSIV Leakage Control Line	Drywell
DBB-1-SS-1	MSIV Leakage Control Line	Drywell
DBB-2-SS-1	MSIV Leakage Control Line	Drywell
DBB-4-SS-1	MSIV Leakage Control Line	Drywell
1	Recirc	Drywell
2	Temp Equal Column "B"	Drywell
3	Temp Equal Column "B"	Drywell
4	Reactor Water Level	Drywell
5	Main Steam	Drywell
6	Main Steam	Drywell
7	Recirc	Drywell
8	Recirc	Drywell
9	Recirc	Drywell
10	RCIC	Drywell
11	RCIC	Drywell
12	RCIC	Drywell
13	Recirc	Drywell
14	Recirc	Drywell
15	Recirc	Drywell
16	Recirc	Drywell
17	Recirc	Drywell
18	Recirc	Drywell
19	Recirc	Drywell
20	Temp Equal Column "A"	Drywell
21	Temp Equal Column "A"	Drywell
22	Recirc	Drywell
1A	Main Steam	Drywell
2A	Main Steam	Drywell
3A	Main Steam	Drywell

*(2 ea) - Indicates there are 2 snubbers with that number

TABLE 4.6-4 (cont)

Identification No.	System	Location
4A	Main Steam	Drywell
5A	Recirc	Drywell
6A	Recirc	Drywell
7A	Recirc	Drywell
8A	Recirc	Drywell
9A	Recirc	Drywell
10A	Recirc	Drywell
11A	Recirc	Drywell
12A	Recirc	Drywell
13A	Recirc	Drywell
14A	Recirc	Drywell
15A	Main Steam	Drywell
16A	Reactor Vessel Vent Line	Drywell
17A	Reactor Vessel Vent Line	Drywell
18A	Reactor Vessel Vent Line	Drywell
19A	Reactor Vessel Vent Line	Drywell
20A	Recirc	Drywell
1B	Reactor Vessel Vent Line	Drywell
2B	Reactor Vessel Vent Line	Drywell
5B	Recirc	Drywell
6B	RCIC	Drywell

Modifications to this Table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

TABLE 4.6-5

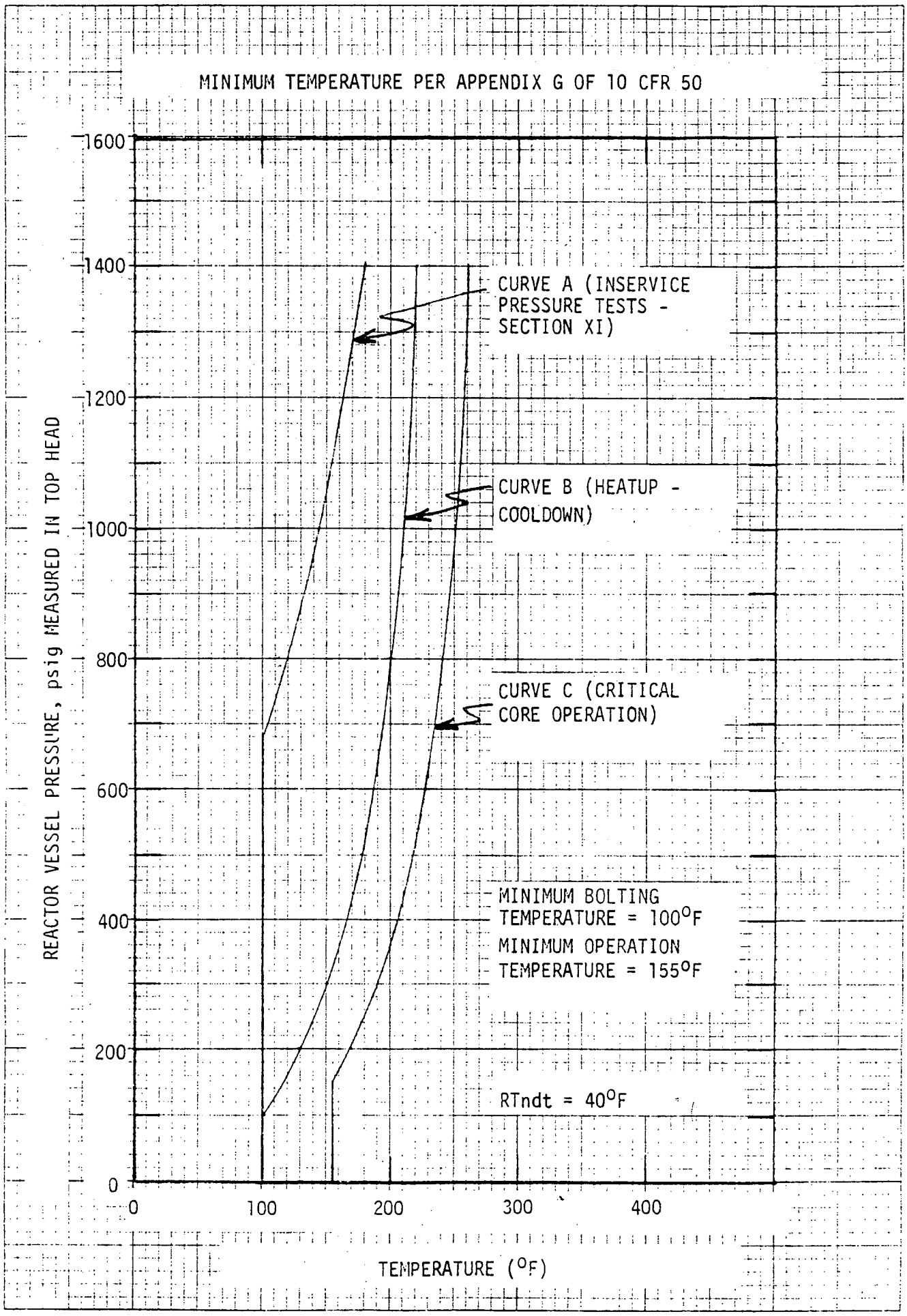
SNUBBERS IN HIGH RADIATION AREA DURING SHUTDOWN

AND/OR ESPECIALLY DIFFICULT TO REMOVE

None

Modifications to this Table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

FIGURE 3.6-1


MINIMUM TEMPERATURE PER APPENDIX G OF 10 CFR 50
CURVE A (INSERVICE PRESSURE TESTS - SECTION XI)
CURVE B (HEATUP - COOLDOWN)
CURVE C (CRITICAL CORE OPERATION)
MINIMUM BOLTING TEMPERATURE = 100°F
MINIMUM OPERATION TEMPERATURE = 155°F
RTndt = 40°F
REACTOR VESSEL PRESSURE, psig MEASURED IN TOP HEAD
TEMPERATURE (°F)
0
200
400
600
800
1000
1200
1400
1600
0
100
200
300
400