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

February 27, 1985

Docket Nos. 50-237/249
LS05-85-02-018

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

SUBJECT: TECHNICAL SPECIFICATION CHANGES RELATING TO SNUBBERS AND REFLECTING THE GUIDANCE OF GENERIC LETTER 84-13

Re: Dresden Nuclear Power Station, Unit Nos. 2 and 3

The Commission has issued the enclosed Amendment No. 85 to Provisional Operating License No. DPR-19 for Dresden Unit 2 and Amendment No. 78 to Facility Operating License No. DPR-25 for Dresden Unit 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated March 15, 1984 as supplemented by your letter dated September 21, 1984.

Following the submittal of your March 15, 1984 application, which requested amendments to update the TS snubber tables to reflect additions, deletions and other changes which had occurred, the Commission issued Generic Letter (GL) 84-13 on May 13, 1984. This GL updated the TS requirements for snubbers and, among other things, allowed licensees to choose to request a license amendment to delete the tabular listing of snubbers from its TS. You made that choice and formalized it in your submittal dated September 21, 1984. This submittal requested authorization for amendments which allow the deletion of TS Tables 3.6.1.a and 3.6.1.b and all references to them, the revision of TS Sections 3.6.I.1 and 3.6.I.4 and their Bases to remove any reference to the Torus Ring Header Snubber work which has already been completed for both units, the revision of TS Section 4.6.2 and the Bases for Section 3.6.I to remove limits on the type of functional testing performed on the snubbers and the revision of TS Sections 4.6.I.2 and 4.6.I.4 and the Bases for 3.6.I to allow for velocity range tests as required for certain types of snubbers which were not used at the site until recently.

The Staff performed an analysis of your amendment request as it relates to the guidance provided in GL 84-13 as well as an evaluation of the requests for removal of limits on the type of functional testing performed on the snubbers and for allowance for velocity range tests as required by certain types of snubbers not used at the site until recently. Based on this, the staff finds the requested changes fall within the guidance of GL 84-13 and within other staff technical positions and are, therefore, acceptable.

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The amendments involve a change in installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents 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 amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet 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 amendments.

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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

A Notice of Consideration of Issuance of Amendments to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested actions in the March 15, 1984 and September 21, 1984 letters was published in the Federal Register on December 31, 1984 (49 FR 50800). No public comments or requests for hearing were received.

This action will appear in the Commission's monthly notice publication in the Federal Register.

Sincerely,

Original signed by

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 85 to License No. DPR-19
2. Amendment No. 78 to License No. DPR-25

cc w/enclosures:
See next page

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 85
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated March 15, 1984 as supplemented by a letter dated September 21, 1984 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 3.B of Provisional Operating License No. DPR-19 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 85 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 27, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 85

PROVISIONAL OPERATING LICENSE DPR-19

DOCKET NO. 50-237

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
i	i
iii(1)	iii(2)
vii(1)	vii(2)
3/4.6-12	3/4.6-12
3/4.6-13	3/4.6-13
3/4.6-15 through 3/4.6-24	3/4.6-15 through 3/4.6-20
B3/4.6-25 through B3/4.6-38	B3/4.6-21 through B3/4.6-34

(1) Pages issued by Amendment 83 (RETS).
(2) Pages which supersede those issued by Amendment 83 (Amd. 83 does not become effective until March 15, 1985).

APPENDIX A

TO

OPERATING LICENSE DPR-19

TECHNICAL SPECIFICATIONS

AND BASES

FOR

DRESDEN NUCLEAR POWER STATION UNIT 2

GRUNDY COUNTY, ILLINOIS

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

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List of Tables

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

2. If specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

3. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

4. Whenever one pump is operable and the remaining pump is in the tripped position, the operable pump shall be at a speed less than 65% before starting the inoperable pump.

I. Snubbers (Shock Suppressors)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

Recirculation pumps speed shall be checked daily for mismatch.

I. Snubbers (Shock) Suppressors)

The following surveillance requirements apply to safety related snubbers.

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. During all modes of operation except cold shutdown and refuel, all safety related snubbers shall be operable except as noted in Specification 3.6.I.2 through 3.6.I.4.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

1. Visual Inspection

An independent visual inspection shall be performed on the safety related hydraulic and mechanical snubbers in accordance with the schedule below.

a. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connection to the piping and anchor to verify snubber operability.

b. All mechanical snubbers shall be visually inspected. This

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

2. Functional Testing

a. Once each refueling cycle, a representative sample of approximately 10% of the hydraulic snubbers shall be functionally tested for operability, including:

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

(ii) Snubber bleed, or release rate, where required, is within the specified range in compression or tension.

3688a

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

For each unit and subsequent unit found inoperable, an additional 10% of the hydraulic snubbers shall be tested until no more failures are found or all units have been tested.

b. Once each refueling cycle, a representative sample of approximately 10% of the mechanical snubbers shall be functionally tested for operability. The test shall consist of two parts:

(i) Verification that the force that initiates free movement of the snubber in either tension or compression is less than the specified maximum breakaway friction force.

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

(ii) Verify that the activation (restraining action) is achieved within the specified range of acceleration or velocity, as applicable based on snubber design in both tension and compression.

For each unit and subsequent unit found inoperable, an additional 10% of the mechanical snubbers shall be so tested until no more failures are found or all units have been tested.

c. 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 resampling.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

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

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When a snubber is is deemed inoperable, a review of all pertinent facts shall be conducted to determine the snubber mode of failure and to decide if an engineering evaluation should be performed on the supported system or components. If said evaluation is deemed necessary, it will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

4. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and, if determined to be a generic deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

3.6 LIMITING CONDITION FOR OPERATION
(Cont'd.)

5. Snubbers may be added or removed from safety related systems without prior license amendment.

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

5. Snubber service life monitoring shall be followed by existing station record systems, including the central filing system, maintenance files, safety related work packages, and snubber inspection records. The above record retention methods shall be used to prevent the hydraulic snubbers from exceeding a service life of 10 years and the mechanical snubbers from exceeding a service life of 40 years (lifetime of the plant).

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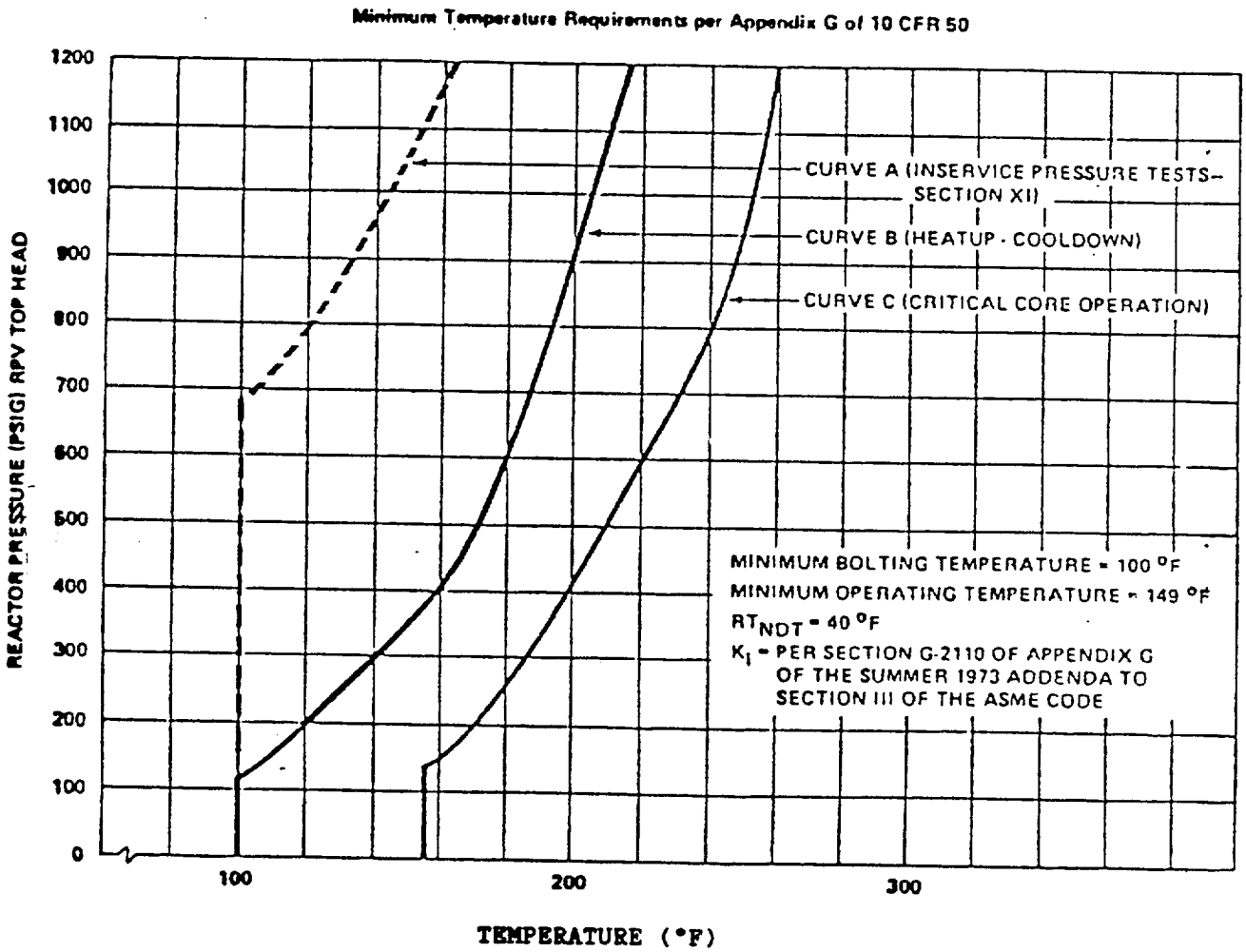

Fig. 3.6.1
MINIMUM TEMPERATURE REQUIREMENTS PER APPENDIX G OF 10 CFR 50

3.6 LIMITING CONDITION FOR OPERATION BASES

A. Thermal Limitations - The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 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 periods for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

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 550°F. Because of the slow temperature-time response of the massive flanges relative to the adjacent head and shell sections, calculated temperatures obtained were 500°F (shell) and 360°F (flange) (140°F differential). Both axial and radial thermal stresses were considered to act concurrently with full primary loadings. 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.

The flange metal temperature differential of 140°F occurred as a result of sluggish temperature response and the fact that the heating rate continued over a 450°F coolant temperature range.

The uncontrolled cooldown rate of 240°F was based on the maximum expected transient over the lifetime of the reactor vessel. This maximum expected transient is the injection of cold water into the vessel by the high pressure coolant injection subsystem. This transient was considered in the design of the pressure vessel and five such cycles were considered in the design. Detailed stress analyses were conducted to assure that the injection of cold water into the vessel by the HPCI would not exceed ASME stress code limitations.

B. Specification 3.6.A.4 increases margin of safety for thermal-hydraulic stability and startup of recirculation pump.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Pressurization Temperature - 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. These restrictions on inservice hydrostatic testing, on heatup and cooldown, and on critical core operation shown in Figure 3.6.1, were established to be in conformance with Appendix G to 10 CFR 50.

In evaluating the adequacy of ferritic steels Sa302B it is necessary that the following be established:

a) The reference nil-ductility temperature (RT_{NDT}) for all vessel and adjoining materials,

b) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev), and

c) the fluence at the location of a postulated flow.

The initial RT_{NDT} of the main closure flange, the shell and head materials connecting to these flanges, and connecting welds is 10°F. However, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. (Reference Appendix F to the FSAR) The closure flanges and connecting shell materials are not subject to any appreciable neutron radiation exposure, nor are the vertical electroslag seams. The flange area is moderately stressed by tensioning the head bolts. Therefore, as is indicated in curves (a) and (b) of Figure 3.6.1, the minimum temperature of the vessel shell immediately below the vessel flange is established as 100°F below a pressure of 400 psig. (40°F + 60°F, where 40°F is the RT_{NDT} of the electroslag weld and 60°F is a conservatism required by the ASME Code). Above approximately 400 psig pressure, the stresses associated with pressurization are more limiting than the bolting stresses, a fact that is reflected in the non-linear portion of curves (a) and (b). Curve (c), which defines the temperature limitations for critical core operation, was established per Section IV 2.c. of Appendix G of 10CFR50. Each of the curves, (a), (b) and (c) define temperature limitations for unirradiated ferric steels. Provision has been made for the modification of these curves to account for the change in RT_{NDT} as a result of neutron embrittlement.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The withdrawal schedule in Table 4.6.2 is based on the three capsule surveillance program as defined in Section 11.C.3.a of 10 CFR 50 Appendix H. The accelerated capsule (Near Core Top Guide) is not required by Appendix H but will be tested to provide additional information on the vessel material.

This surveillance program conforms to ASTM E 185-73 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" with one exception. The base metal specimens of the vessel were made with their longitudinal axes parallel to the principal rolling direction of the vessel plate.

C. Coolant Chemistry - A radioactivity concentration limit of 20 Micro-Ci/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Specification 3.8.C.1 or there is a failure or a prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture, outside the drywell, the resultant radiological dose at the site boundary would be about 10 rem to the thyroid. This dose was calculated on the basis of a total iodine activity limit of 20 Micro-Ci/ml, meteorology corresponding to Type F conditions with a one meter per second wind speed, and a valve closure time of five seconds. If the valve closed in ten seconds, then the resultant dose would increase to about 25 rem.

The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hours. In addition, the trend of the stack off-gas release rate, which is continuously monitored, is a good indicator 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, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam lines.

Materials in the primary system are primarily 304 stainless steel and the Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

the stainless steel. The attached graph, Figure 4.6.2, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup and hot standby. During these periods with steaming rates less than 100,000 pounds per hour, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Figure 4.6.2 are not exceeded. At steaming rates of at least 100,000 pounds per hour, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if 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; e.g., Na_2SO_4, 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 clean-up 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 clean-up system to re-establish the purity of the reactor coolant.

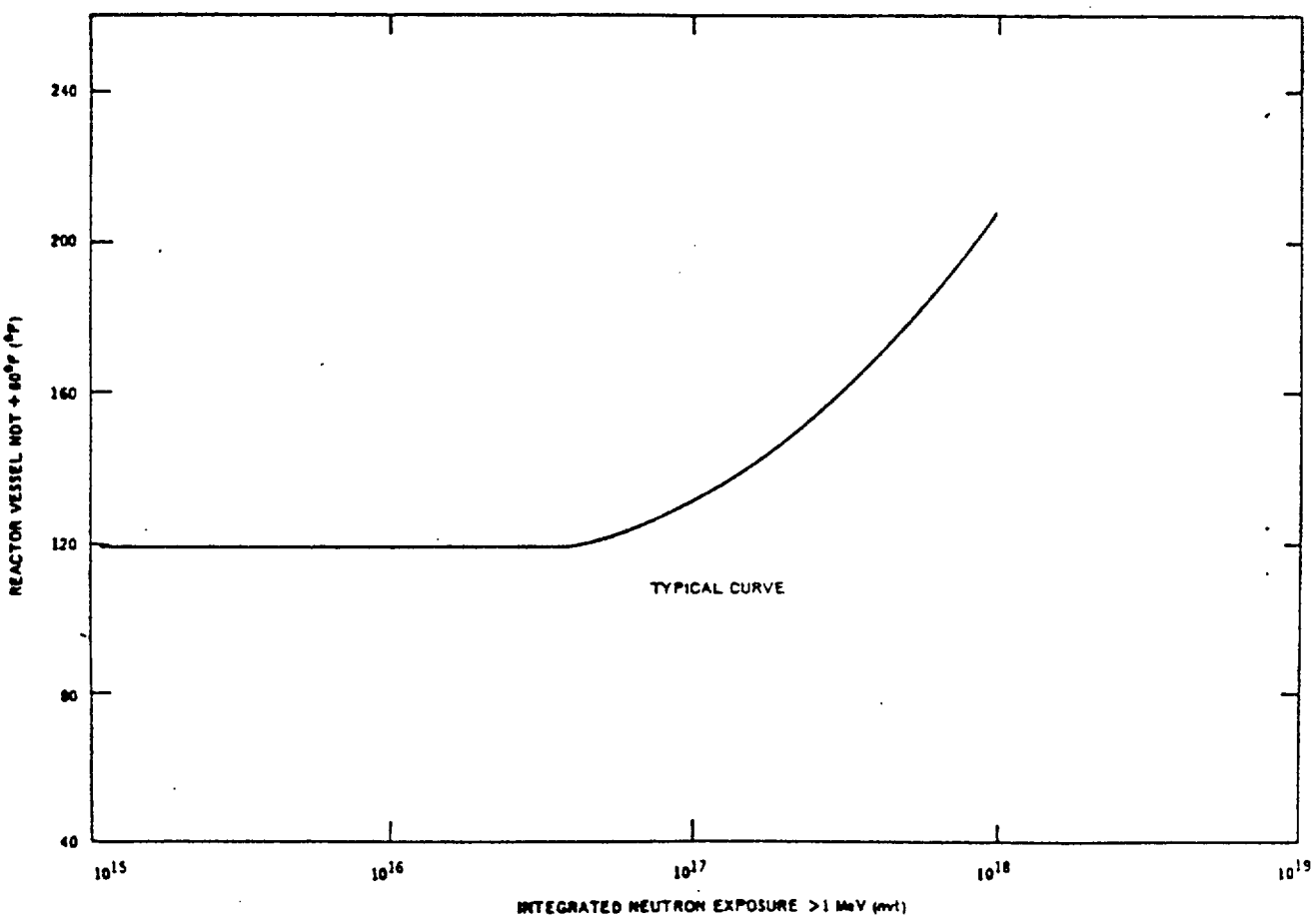

Figure 4.6.1

MINIMUM REACTOR PRESSURIZATION TEMPERATURE

TABLE 4.6.2

NEUTRON FLUX AND SAMPLES WITHDRAWAL
SCHEDULE FOR DRESDEN UNIT 2

Withdrawal Year	Part No.	Location	Comments
1977	6	Near Core Top Guide - 180°	Accelerated Sample
1980	8	Wall - 215°	
2000	7	Wall - 95°	
	9	Wall - 245°	Standby
	10	Wall - 275°	Standby

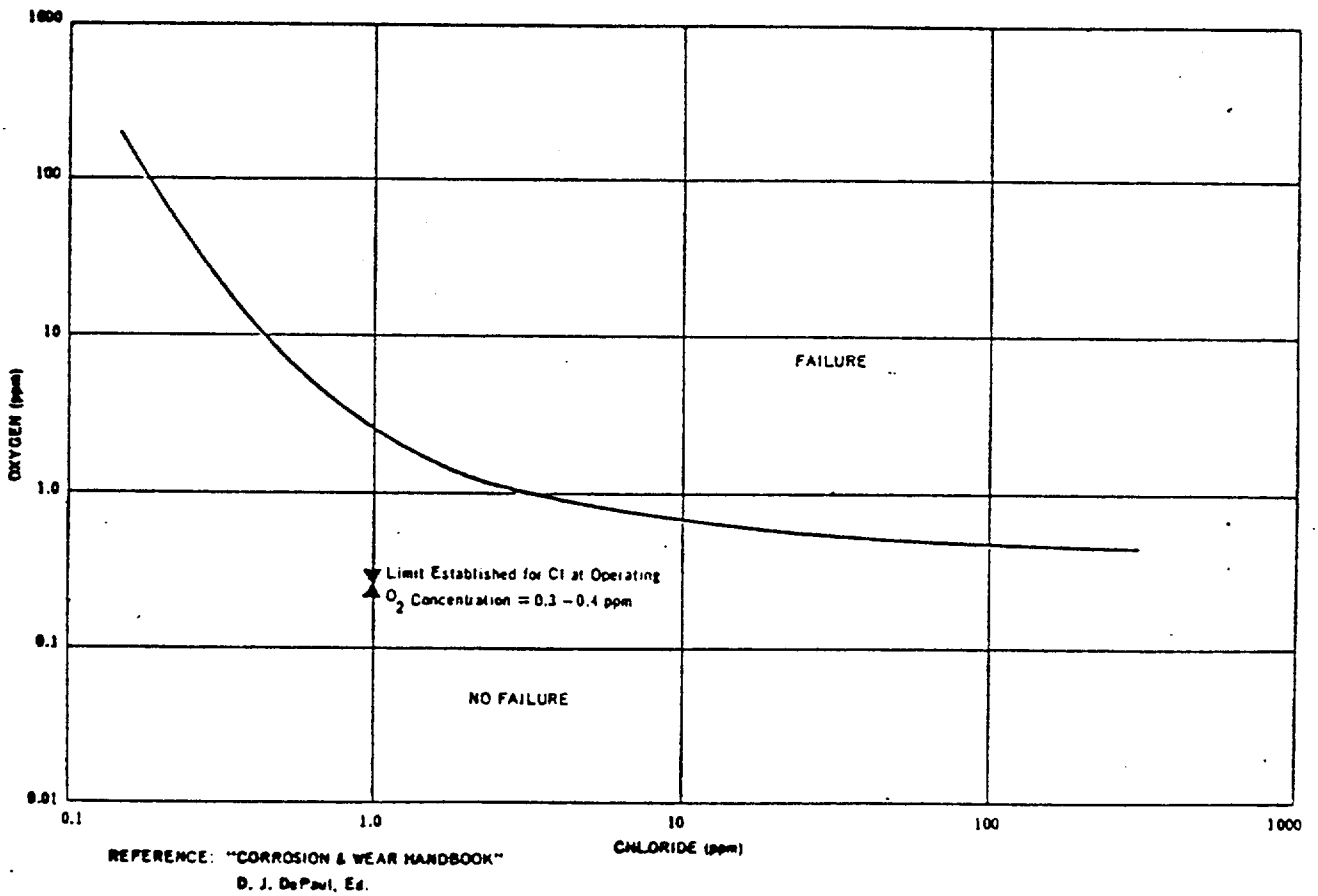

Figure 4.6.2

CHLORIDE STRESS CORROSION TEST RESULTS AT 500°F

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

During start-up periods, which are in the category of less than 100,000 pounds per hour, conductivity may exceed 2 micro-mho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 micro-mho (other than short term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. 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 required by Specification 4.6.C.3 may be performed by a gamma scan.

D. 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 makeup 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.D 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.D, 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

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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 additional leakage requirements will be in effect only while the reactor is operated with the recirculation flaws detected during the 1983 Refueling Outage. The additional leakage requirements will provide more conservative detection and corrective action should the current flaws propagate thru wall.

The capacity of the drywell sump is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

The performance of reactor coolant leakage detection system will be evaluated during the first five years of station operation and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detection system is capable of detecting the order of 3000 lb/hr.

The system performance will be evaluated during the first five years of plant operation and the conclusions of the evaluation will be reported to the NRC.

E. Safety and Relief Valves - The frequency and testing requirements for the safety and relief valves are specified in the Inservice Testing Program which is based on Section XI of the ASME Boiler and Pressure Vessel Code. Adherence to these code requirements provides adequate assurance as to the proper operational readiness of these valves. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as plus or minus 1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher than the reactor coolant pressure safety limit of 1375 psig is not exceeded. The safety valves are required to be operable above the design pressure (90 psig) at which the core spray subsystems are not designed to deliver full flow.

F. Structural integrity - A pre-service inspection of the components in the primary coolant pressure boundary will be conducted after site erection to assure the system is free of gross defects and as a reference base for later inspections. Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout life.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Inservice Inspections of ASME Code Class 1, 2 and 3 components will be performed in accordance with the applicable version of Section XI of the ASME Boiler and Pressure Vessel Code. Relief from any of the above requirements must be provided in writing by the Commission. The Inservice Inspection program and the written relief do not form a part of these Technical Specifications.

These studies show that it requires thousands of stress cycles at stresses beyond any 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 sensitive, fast and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

After five years of operation, a program for in-service inspection of piping and components within the primary pressure boundary which are outside the downstream containment isolation valve shall be submitted to the NRC.

G. 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 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.

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.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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 or broken 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 twenty 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 operatng 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.

H. Recirculation Pump Flow Mismatch

The LPCI loop selection logic has been described in the Dresden Nuclear Power Station Units 2 and 3 FSAR, Amendments 7 and 8. 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 not be expected to function at a speed differential of 15%. Below 80% power, the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

In addition, during the start-up of Dresden Unit 2, it was found that a flow mismatch between the two sets of jet pumps caused by a difference in recirculation loops could set up a vibration until a mismatch in speed of 27% occurred. The 10% and 15% speed mismatch restrictions provide additional margin before a pump vibration problem will occur.

ECCS performance during reactor operation with one recirculation loop out of service has not been analyzed. Therefore, sustained reactor operation under such conditions is not permitted.

I. Snubbers (Shock Suppressors)

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.I.4 prohibits startup with inoperable snubbers.

When a snubber is found inoperable, a review shall be performed to determine the snubber mode of failure. Results of the review shall be used to determine if an engineering evaluation of the safety-related system or component is necessary. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the support component or system.

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

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

All safety related mechanical snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation and attachments to the piping and anchor for indication of damage or impaired operability.

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.

To further increase the assurance of snubber reliability, functional tests will be performed once each refueling cycle. A representative sample of 10% of the safety-related snubbers will be functionally tested. Observed failures on these samples will require testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as different entities for the above surveillance programs.

Hydraulic snubber testing will include stroking of the snubbers to verify piston movement, lock-up, and bleed. Functional testing of the mechanical snubbers will consist of verification that the force that initiates free movement of the snubber in either tension or compression is less than the maximum breakaway friction force and verification that the activation (restraining action) is achieved within the specified range of acceleration or velocity, as applicable based on snubber design, in both tension and compression.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

When the cause of rejection of the 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.

Monitoring of snubber service life shall consist of the existing station record systems, including the central filing system, maintenance files, safety-related work packages, and snubber inspection records. The record retention programs employed at the station shall allow station personnel to maintain snubber integrity. The service life for hydraulic snubbers is 10 years. The hydraulic snubbers existing locations do not impose undue safety implications on the piping and components because they are not exposed to excesses in environmental conditions. The service life for mechanical snubbers is 40 years, lifetime of the plant. The mechanical snubbers are installed in areas of harsh environmental conditions because of their dependability over hydraulic snubbers in these areas. All snubber installations have been thoroughly engineered providing the necessary safety requirements. Evaluations of all snubber locations and environmental conditions justify the above conservative snubber service lives.

4.6 SURVEILLANCE REQUIREMENT BASES

None

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 78
License No. DPR-25

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated March 15, 1984 as supplemented by a letter dated September 21, 1984 complies with 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 3.B of Facility Operating License No. DPR-25 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 78, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 27, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 78

FACILITY OPERATING LICENSE DPR-25

DOCKET NO. 50-249

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
iii(1)	iii(2)
vii(1)	vii((2)
viii(1)	viii(2)
3/4.6-12	3/4.6-12
3/4.6-13	3/4.6-13
3/4.6-15 through 3/4.6-24	3/4.6-15 through 3/4.6-20
B3/4.6-25 thorugh B3/4.6-38	B3/4.6-21 through B3/4.6-34

(1) Pages issued by Amendment 77 (RETS).
(2) Pages which supersede those issued by Amendment 77 (Amd. 77 does not become effective until March 15, 1985).

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

2. If specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

3. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

4. Whenever one pump is operable and the remaining pump is in the tripped position, the operable pump shall be at a speed less than 65% before starting the inoperable pump.

I. Snubbers (Shock Suppressors)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

Recirculation pumps speed shall be checked daily for mismatch.

I. Snubbers (Shock) Suppressors)

The following surveillance requirements apply to safety related snubbers.

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. During all modes of operation except cold shutdown and refuel, all safety related snubbers shall be operable except as noted in Specification 3.6.I.2 through 3.6.I.4.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

1. Visual Inspection

An independent visual inspection shall be performed on the safety related hydraulic and mechanical snubbers in accordance with the schedule below.

a. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connection to the piping and anchor to verify snubber operability.

b. All mechanical snubbers shall be visually inspected. This

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

2. Functional Testing

a. Once each refueling cycle, a representative sample of approximately 10% of the hydraulic snubbers shall be functionally tested for operability, including:

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

(ii) Snubber bleed, or release rate, where required, is within the specified range in compression or tension.

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

For each unit and subsequent unit found inoperable, an additional 10% of the hydraulic snubbers shall be tested until no more failures are found or all units have been tested.

b. Once each refueling cycle, a representative sample of approximately 10% of the mechanical snubbers shall be functionally tested for operability. The test shall consist of two parts:

(i) Verification that the force that initiates free movement of the snubber in either tension or compression is less than the specified maximum breakaway friction force.

3.6 LIMITING CONDITION FOR OPERATION
(Cont'd.)

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

(ii) Verify that the activation (restraining action) is achieved within the specified range of acceleration or velocity, as applicable based on snubber design in both tension and compression.

For each unit and subsequent unit found inoperable, an additional 10% of the mechanical snubbers shall be so tested until no more failures are found or all units have been tested.

c. 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 resampling.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

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

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When a snubber is deemed inoperable, a review of all pertinent facts shall be conducted to determine the snubber mode of failure and to decide if an engineering evaluation should be performed on the supported system or components. If said evaluation is deemed necessary, it will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

4. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and, if determined to be a generic deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

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3.6 LIMITING CONDITION FOR OPERATION
(Cont'd.)

5. Snubbers may be added or removed from safety related systems without prior license amendment.

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

5. Snubber service life monitoring shall be followed by existing station record systems, including the central filing system, maintenance files, safety related work packages, and snubber inspection records. The above record retention methods shall be used to prevent the hydraulic snubbers from exceeding a service life of 10 years and the mechanical snubbers from exceeding a service life of 40 years (lifetime of the plant).

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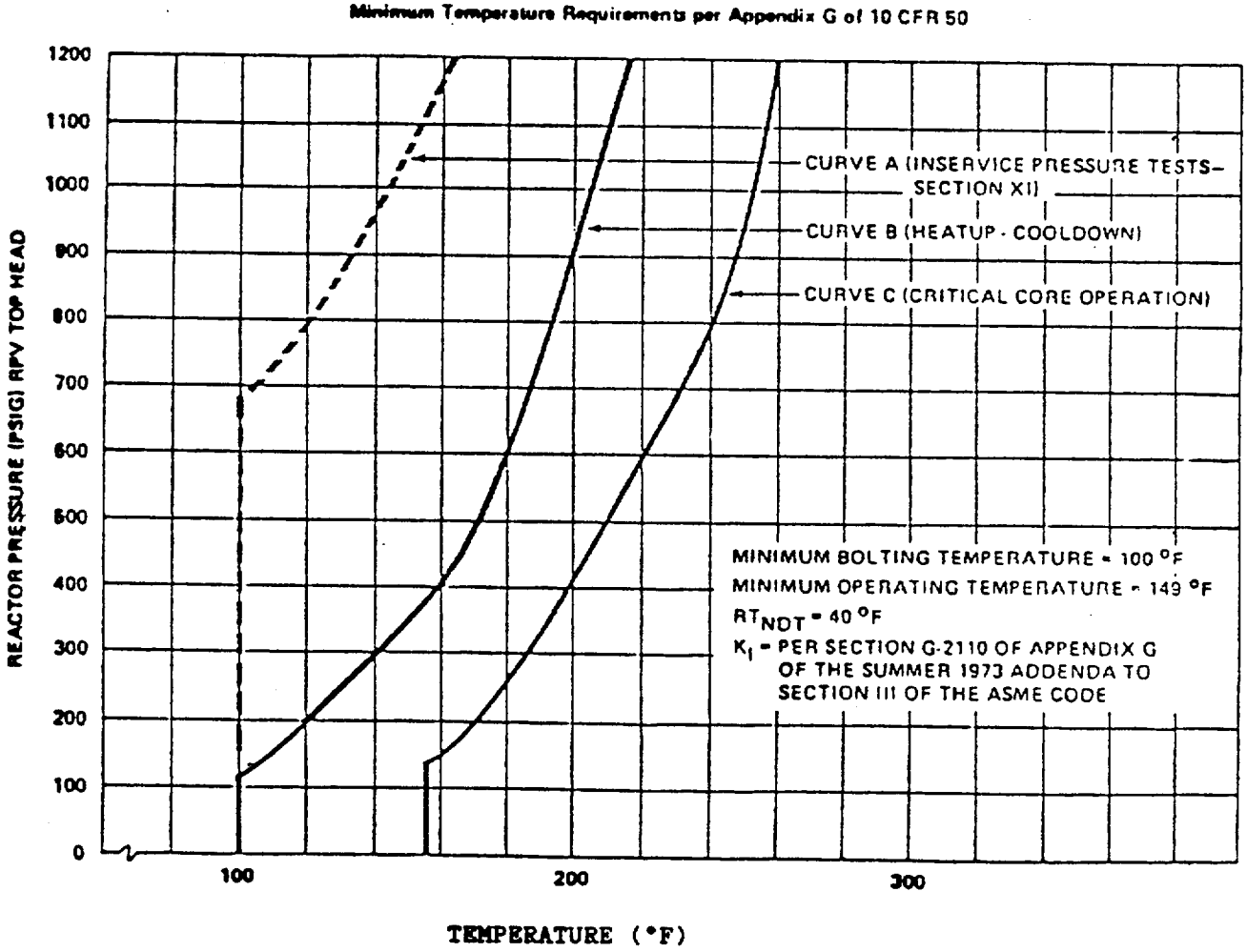

Fig. 3.6.1
MINIMUM TEMPERATURE REQUIREMENTS PER APPENDIX G OF 10 CFR 50

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3.6 LIMITING CONDITION FOR OPERATION BASES

A. Thermal Limitations - The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 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 periods for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

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 550°F. Because of the slow temperature-time response of the massive flanges relative to the adjacent head and shell sections, calculated temperatures obtained were 500°F (shell) and 360°F (flange) (140°F differential). Both axial and radial thermal stresses were considered to act concurrently with full primary loadings. 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.

The flange metal temperature differential of 140°F occurred as a result of sluggish temperature response and the fact that the heating rate continued over a 450°F coolant temperature range.

The uncontrolled cooldown rate of 240°F was based on the maximum expected transient over the lifetime of the reactor vessel. This maximum expected transient is the injection of cold water into the vessel by the high pressure coolant injection subsystem. This transient was considered in the design of the pressure vessel and five such cycles were considered in the design. Detailed stress analyses were conducted to assure that the injection of cold water into the vessel by the HPCI would not exceed ASME stress code limitations.

B. Specification 3.6.A.4 increases margin of safety for thermal-hydraulic stability and startup of recirculation pump.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Pressurization Temperature - 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. These restrictions on inservice hydrostatic testing, on heatup and cooldown, and on critical core operation shown in Figure 3.6.1, were established to be in conformance with Appendix G to 10 CFR 50.

In evaluating the adequacy of ferritic steels Sa302B it is necessary that the following be established:

a) The reference nil-ductility temperature (RT_{NDT}) for all vessel and adjoining materials,

b) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev), and

c) the fluence at the location of a postulated flow.

The initial RT_{NDT} of the main closure flange, the shell and head materials connecting to these flanges, and connecting welds is 10°F. However, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. (Reference Appendix F to the FSAR) The closure flanges and connecting shell materials are not subject to any appreciable neutron radiation exposure, nor are the vertical electroslag seams. The flange area is moderately stressed by tensioning the head bolts. Therefore, as is indicated in curves (a) and (b) of Figure 3.6.1, the minimum temperature of the vessel shell immediately below the vessel flange is established as 100°F below a pressure of 400 psig. (40°F + 60°F, where 40°F is the RT_{NDT} of the electroslag weld and 60°F is a conservatism required by the ASME Code). Above approximately 400 psig pressure, the stresses associated with pressurization are more limiting than the bolting stresses, a fact that is reflected in the non-linear portion of curves (a) and (b). Curve (c), which defines the temperature limitations for critical core operation, was established per Section IV 2.c. of Appendix G of 10CFR50. Each of the curves, (a), (b) and (c) define temperature limitations for unirradiated ferric steels. Provision has been made for the modification of these curves to account for the change in RT_{NDT} as a result of neutron embrittlement.

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3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The withdrawal schedule in Table 4.6.2 is based on the three capsule surveillance program as defined in Section 11.C.3.a of 10 CFR 50 Appendix H. The accelerated capsule (Near Core Top Guide) is not required by Appendix H but will be tested to provide additional information on the vessel material.

This surveillance program conforms to ASTM E 185-73 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" with one exception. The base metal specimens of the vessel were made with their longitudinal axes parallel to the principal rolling direction of the vessel plate.

C. Coolant Chemistry - A radioactivity concentration limit of 20 Micro-Ci/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Specification 3.8.C.1 or there is a failure or a prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture, outside the drywell, the resultant radiological dose at the site boundary would be about 10 rem to the thyroid. This dose was calculated on the basis of a total iodine activity limit of 20 Micro-Ci/ml, meteorology corresponding to Type F conditions with a one meter per second wind speed, and a valve closure time of five seconds. If the valve closed in ten seconds, then the resultant dose would increase to about 25 rem.

The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hours. In addition, the trend of the stack off-gas release rate, which is continuously monitored, is a good indicator 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, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam lines.

Materials in the primary system are primarily 304 stainless steel and the Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

the stainless steel. The attached graph, Figure 4.6.2, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup and hot standby. During these periods with steaming rates less than 100,000 pounds per hour, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Figure 4.6.2 are not exceeded. At steaming rates of at least 100,000 pounds per hour, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if 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; e.g., Na_2SO_4, 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 clean-up 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 clean-up system to re-establish the purity of the reactor coolant.

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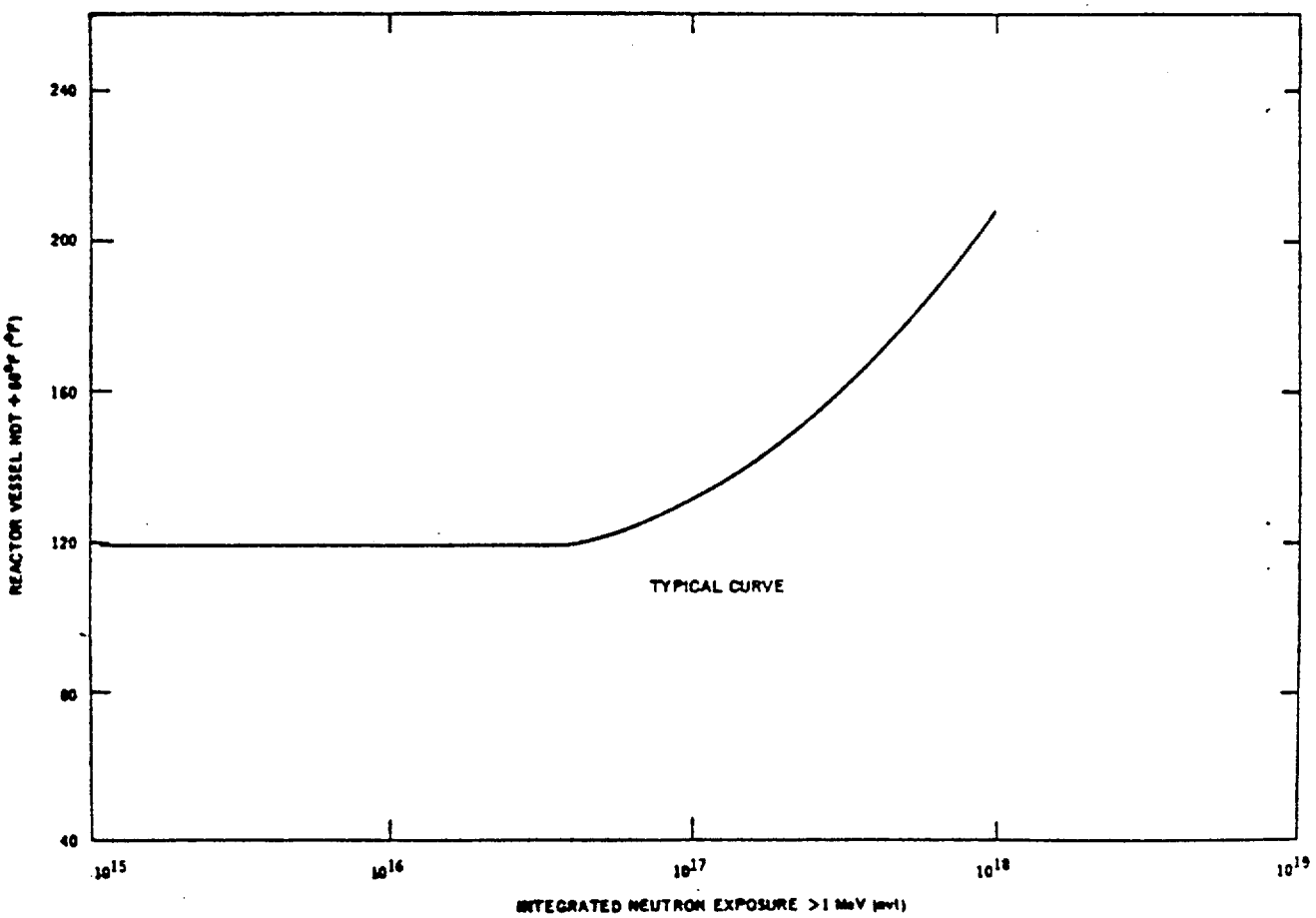

Figure 4.6.1

MINIMUM REACTOR PRESSURIZATION TEMPERATURE

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TABLE 4.6.2

NEUTRON FLUX AND SAMPLES WITHDRAWAL
SCHEDULE FOR DRESDEN UNIT 3

Withdrawal Year	Part No.	Location	Comments
1978	16	Near Core Top Guide - 180°	Accelerated
1981	18	Wall - 215°	
2001	19	Wall - 245°	
	15	Wall - 65°	Standby
	20	Wall - 275°	Standby

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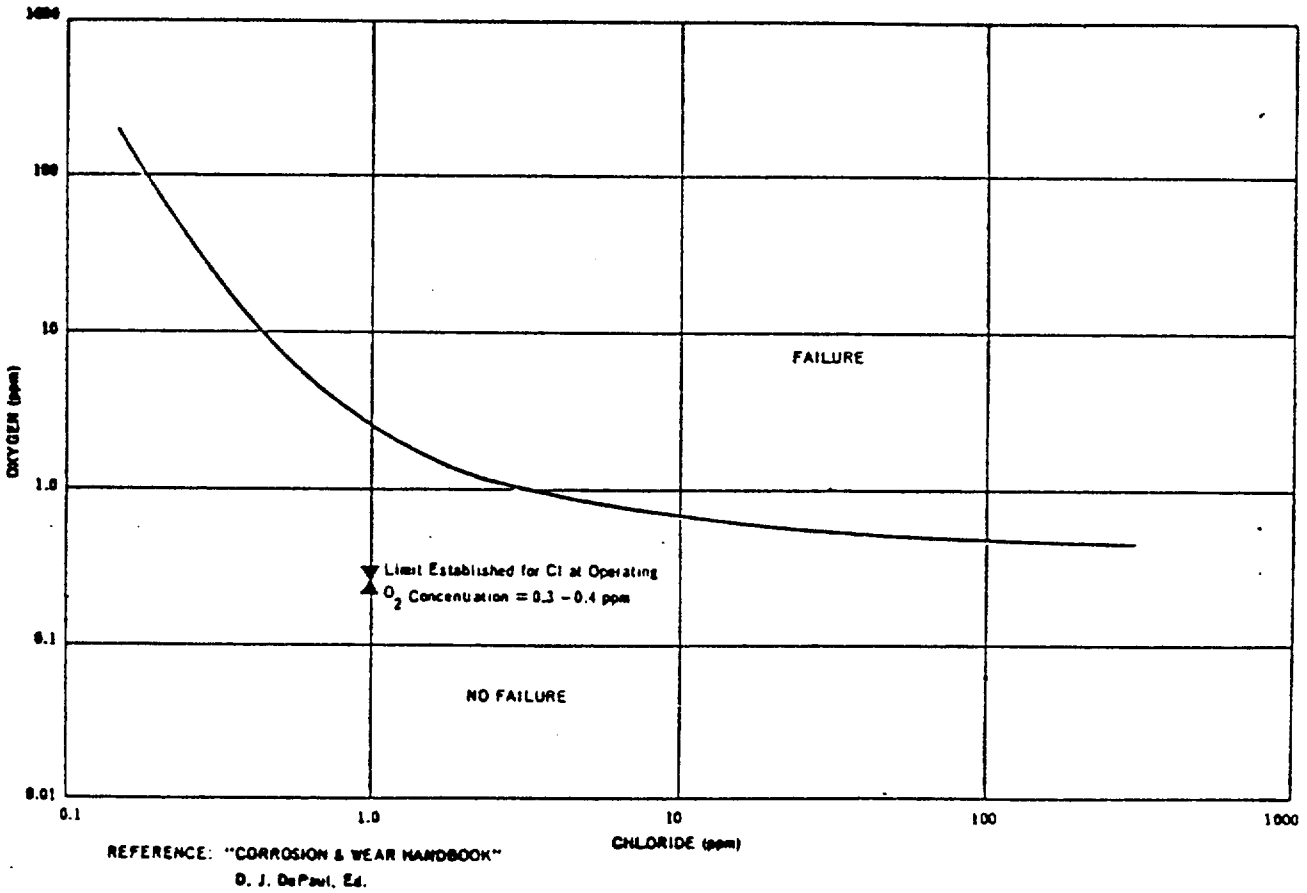

Figure 4.6.2

CHLORIDE STRESS CORROSION TEST RESULTS AT 500°F

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3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

During start-up periods, which are in the category of less than 100,000 pounds per hour, conductivity may exceed 2 micro-mho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 micro-mho (other than short term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. 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 required by Specification 4.6.C.3 may be performed by a gamma scan.

D. 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 makeup 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.D 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.D, 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

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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 capacity of the drywell sump is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

The performance of reactor coolant leakage detection system will be evaluated during the first five years of station operation and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detection system is capable of detecting the order of 3000 lb/hr.

The system performance will be evaluated during the first five years of plant operation and the conclusions of the evaluation will be reported to the NRC.

E. Safety and Relief Valves - The frequency and testing requirements for the safety and relief valves are specified in the Inservice Testing Program which is based on Section XI of the ASME Boiler and Pressure Vessel Code. Adherence to these code requirements provides adequate assurance as to the proper operational readiness of these valves. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as plus or minus 1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher than the reactor coolant pressure safety limit of 1375 psig is not exceeded. The safety valves are required to be operable above the design pressure (90 psig) at which the core spray subsystems are not designed to deliver full flow.

F. Structural integrity - A pre-service inspection of the components in the primary coolant pressure boundary will be conducted after site erection to assure the system is free of gross defects and as a reference base for later inspections. Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout life.

Inservice Inspections of ASME Code Class 1, 2 and 3 components will be performed in accordance with the applicable version of Section XI of the ASME Boiler and Pressure Vessel Code.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Relief from any of the above requirements must be provided in writing by the Commission. The Inservice Inspection program and the written relief do not form a part of these Technical Specifications.

These studies show that it requires thousands of stress cycles at stresses beyond any 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 sensitive, fast and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

After five years of operation, a program for in-service inspection of piping and components within the primary pressure boundary which are outside the downstream containment isolation valve shall be submitted to the NRC.

G. 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 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.

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.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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 or broken 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 twenty 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 operatng 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.

H. Recirculation Pump Flow Mismatch

The LPCI loop selection logic has been described in the Dresden Nuclear Power Station Units 2 and 3 FSAR, Amendments 7 and 8. 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 not be expected to function at a speed differential of 15%. Below 80% power, the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

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3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

In addition, during the start-up of Dresden Unit 2, it was found that a flow mismatch between the two sets of jet pumps caused by a difference in recirculation loops could set up a vibration until a mismatch in speed of 27% occurred. The 10% and 15% speed mismatch restrictions provide additional margin before a pump vibration problem will occur.

ECCS performance during reactor operation with one recirculation loop out of service has not been analyzed. Therefore, sustained reactor operation under such conditions is not permitted.

I. Snubbers (Shock Suppressors)

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.I.4 prohibits startup with inoperable snubbers.

When a snubber is found inoperable, a review shall be performed to determine the snubber mode of failure. Results of the review shall be used to determine if an engineering evaluation of the safety-related system or component is necessary. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the support component or system.

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

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

All safety related mechanical snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation and attachments to the piping and anchor for indication of damage or impaired operability.

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.

To further increase the assurance of snubber reliability, functional tests will be performed once each refueling cycle. A representative sample of 10% of the safety-related snubbers will be functionally tested. Observed failures on these samples will require testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as different entities for the above surveillance programs.

Hydraulic snubber testing will include stroking of the snubbers to verify piston movement, lock-up, and bleed. Functional testing of the mechanical snubbers will consist of verification that the force that initiates free movement of the snubber in either tension or compression is less than the maximum breakaway friction force and verification that the activation (restraining action) is achieved within the specified range of acceleration or velocity, as applicable based on snubber design, in both tension and compression.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

When the cause of rejection of the 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.

Monitoring of snubber service life shall consist of the existing station record systems, including the central filing system, maintenance files, safety-related work packages, and snubber inspection records. The record retention programs employed at the station shall allow station personnel to maintain snubber integrity. The service life for hydraulic snubbers is 10 years. The hydraulic snubbers existing locations do not impose undue safety implications on the piping and components because they are not exposed to excesses in environmental conditions. The service life for mechanical snubbers is 40 years, lifetime of the plant. The mechanical snubbers are installed in areas of harsh environmental conditions because of their dependability over hydraulic snubbers in these areas. All snubber installations have been thoroughly engineered providing the necessary safety requirements. Evaluations of all snubber locations and environmental conditions justify the above conservative snubber service lives.

4.6 SURVEILLANCE REQUIREMENT BASES

None

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