ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES

DRESDEN UNIT 2 - DPR-19

REVISED PAGES	DESCRIPTION
Pg. 3/4.6-5, Para. 3.6.D.1	Revise coolant leakage limits per Generic Letter 84-11 Attachment 1, Item B. Change the reference to "irradiated fuel being in the vessel and coolant temperature above 212°F," to "whenever the reactor is at operating pressure."
Pg. 3/4.6-6, Para. 3.6.D.2.a	New paragraph to include the operability requirements for the primary containment sump collection and flow monitoring system.
Pg. 3/4.6-6, Para. 3.6.D.2.b	New paragraph for primary containment sump sampling and air sampling operability requirements.
Pg. 3/4.6-6, Para. 3.6.D.2 and 4.6.D.2	Delete paragraphs - replaced by the above changes.
Pg. 3/4.6-29, Limiting Condition for Operation Bases(D)	Delete the second paragraph referring to additional leakage requirements as they are no longer required.
Pg. 3/4.6-8, Para. 4.6.F.1	Change the ISI program time period from 40 months to 120 months to reflect the second 10-year period per 10 CFR 50.55a.
Pg. 3/4.1-7, Note **	Typographical error. Change 56% to 50%.
Pg. 3/4.2-20, Table 4.2.1	Typographical error. Under HPCI isolation, change Note (12) to Note (13) for: 1. Steam line high flow. Change note (13) to Note (12) for: 3. Low reactor pressure.
Pg. 3/4.6-7 through 3/4.6-15	Renumbered pages.

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

hours after placing the reactor in the power operating condition.

4. Except as specified in 3.6.C.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hour:
Conductivity 5 micro-mho/cm
Chloride ion 0.5 ppm

5. If Specification 3.6.C.1, 3.6.C.2, 3.6.C.3 or 3.6.C.4 is not met, an orderly shutdown shall be initiated.

D. Coolant Leakage

1. Whenever the reactor is at operating pressure the following will apply to reactor coolant leakage into the primary containment:

a. Unidentified leakage will be limited to a 2 gpm increase within any 24 hour period except that 24 hour period after

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

monitors indicate abnormal conductivity (other than short-term spikes) and analyzed for conductivity and chloride ion content.

b. When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least daily and analyzed for conductivity and chloride ion content.

D. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump flow monitoring and air sampling systems. Sump flow monitoring and recording shall be performed once per 4 hours. Air sampling shall be performed once per day.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

reaching Reactor Power Operation.

b. Unidentified leakage shall not exceed 5 gpm.

c. Total leakage shall not exceed 25 gpm.

If any of the above conditions cannot be met, an orderly shutdown shall be initiated and the Reactor shall be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

2. a. The primary containment floor drain and equipment drain sump flow monitoring systems shall be operable whenever the Reactor is at operating pressure. With either of these systems inoperable, restore the inoperable system to operable status within 24 hours or immediately initiate an orderly shutdown and be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

b. The primary containment sump sampling system and an air sampling system shall be operable during power operation. If either a sump water sample or a containment air sample cannot be obtained for any reason, Reactor operation is permissible only during the succeeding seven days unless the system is made operable during this period.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

E. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320°F, all nine of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be less than or equal to 90 psig and less than or equal to 320° F within 24 hours.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (Psig)
1	1135*
2	1240
2	1250
2	1260
2	1260

The allowable set point error for each valve is plus or minus 1% (FSAR Table 4.5.2:1)

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Valve No.	Set Point (psig)
203-3A	1124*
203-3B	1101
203-3C	1101
203-3D	1124
203-3E	1124

* Target Rock combination safety/relief valve

The allowable setpoint error for each valve is plus or minus 1%.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components".

Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw

indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose inservice examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

F. Structural Integrity

1. Beginning March 1, 1982, and updated every 120 months thereafter, the component inservice inspection program shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been given by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

requirements are met:

(i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI.

(ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

b. For components approved for continued service in accordance with paragraph "a" above, reexamination of the area containing the

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

flaw indication(s) shall be conducted during each scheduled successive inservice inspection. An analysis and evaluation shall be submitted to the NRC following each inservice inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each inservice inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

c. Repair or replacement of components, including reexaminations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 <u>LIMITING CONDITION FOR OPERATION</u> (Cont'd.)

G. Jet Pumps

1. Whenever the Reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

4.6 <u>SURVEILLANCE REQUIREMENT</u> (Cont'd.)

G. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the Startup/Hot Standby or Run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of any jet pumps in the idle loop shall not vary by more than 10% from established patterns.

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

H. Recirculation Pump Flow Mismatch

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

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3. The baseline data required to evaluate the conditions in Specifications 4.6.G.1 and 4.6.G.2 will be acquired each operating cycle.

H. Recirculation Pump Flow Mismatch

Recirculation pumps speed shall be checked daily for mismatch.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

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)

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.)

I. Snubbers (Shock) Suppressors)

The following surveillance requirements apply to safety related snubbers.

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

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

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 inspection shall consist of, but not necessarily be limited to, inspection of the snubber and attachments to the piping and anchor for indications of damage or impaired operability.

No. of Snubbers Found Inoperable During Inspection Interval	Next Required Inspection Interval
0	18 months plus or minus 25%
1	12 months plus or minus 25%
2	6 months plus or minus 25%
3,4	124 days plus or minus 25%
5,6,7	62 days plus or minus 25%
8 or more	31 days plus or minus 25%

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.)

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

Snubbers may be categorized in two groups, "accessible" or "inaccessible," based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

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.

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 pump 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.

NOTES: (For Table 3.1.1 Cont'd.)

14. The APRM 15% scram is bypassed in the run mode.

15. Due to addition of hydrogen to the primary coolant, the Main Steam Line Radiation monitor setting will be less than or equal to 3 times full power background without hydrogen addition for all conditions except for greater than 20% power with hydrogen being injected during which the Main Steam Line Radiation trip setting will be less than or equal to 3 times full power background with hydrogen addition. Required changes in Main Steam Line Radiation Monitor trip setting will be made within 24 hrs. except during controlled power descensions at which time the setpoint change will be made prior to going below 20% power. If due to a recirculation pump trip or other unanticipated power reduction event the reactor is below 20% power without the setpoint change, control rod motion will be suspended until the necessary trip setpoint adjustment is made.

* If the first column cannot be met for one of the trip systems, that trip system shall be tripped.

If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

a. Initiate insertion of operable rods and complete insertion of all operable rods within 4 hours.

b. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

c. Reduce turbine load and close main steam line isolation valves within 5 hours.

d. In the refuel mode, when any control rod is withdrawn, suspend all operations involving core alterations and insert all insertable control rods within one hour.

** An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM's to an APRM.

*** 1 inch on the water level instrumentation is greater than or equal to 504" above vessel zero (see Bases 3.2).

**** Trips upon actuation of the fast closure solenoid which trips the turbine control valves.

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS INSTRUMENTATION, ROD BLOCKS, AND ISOLATIONS

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
Main Steam Line Isolation			
1. Steam Tunnel High Temperature	Refueling Outage	Refuel Outage	None
2. Steam Line High Flow	(1)	Once/3 Months	Once/Day
3. Steam Line Low Pressure	(1)	Once/3 Months	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months (4)	Once/Day
Isolation Condenser Isolation			
1. Steam Line High Flow	(1)	Once/3 Months	None
2. Condensate Line High Flow	(1)	Once/3 Months	None
HPCI Isolation			
1. Steam Line High Flow	(1) (11) (13)	(11) (13)	None
2. Steam Line Area High Temperature	Refueling Outage	Refuel Outage	None
3. Low Reactor Pressure	(1) (12)	(12)	None
Reactor Building Vent Isolation and SBGTS Initiation			
1. Refueling Floor Radiation Monitors	(1)	Once/3 Months	Once/Day

NOTES: (For Table 4.2.1)

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Dresden Unit 2.

2. Function test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. See Note 4.

4. These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed during each refueling outage.

5. A minimum of two channels is required.

(Cont'd. next page)

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES

DRESDEN UNIT 3 - DPR-19

REVISED PAGES	DESCRIPTION
Pg. 3/4.6-7, Para. 4.6.F.1	Change the ISI program time period from 40 months to 120 months to reflect the second 10-year period per 10 CFR 50.55a.

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be less than or equal to 90 psig and less than or equal to 320° F within 24 hours.

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components".

Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

The allowable set point error for each valve is plus or minus 1%

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Valve No.	Set Point (psig)
203-3A	1124*
203-3B	1101
203-3C	1101
203-3D	1124
203-3E	1124

* Target Rock combination safety/relief valve

The allowable setpoint error for each valve is plus or minus 1%.

F. Structural Integrity

1. Beginning March 1, 1982, and updated every 120 months thereafter, the component inservice inspection program shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been given by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

ATTACHMENT 3

SIGNIFICANT HAZARDS CONSIDERATION

DESCRIPTION OF AMENDMENT REQUEST

The proposed changes to the Technical Specifications for Dresden Station Units 2 and 3 are as follows:

(1) The coolant leakage limits and sump pump operability requirements for Unit 2 have been revised to reflect the requirements of Generic Letter 84-11, Attachment 1, Section B. The change reflects a more stringent limit on coolant leakage increases (2 gpm in 24 hours versus existing 1 gpm in 4 hours) and requires plant shutdown if the limit is exceeded regardless of the source of leakage. Operability requirements for related equipment are also added.

(2) The time period for the inservice inspection (ISI) program has been modified to reflect the second 10-year ISI period for Units 2 and 3. This change is administrative in that it merely reflects that the Station has entered the second 10-year ISI period as defined in 10 CFR 50.55a(g).

(3) Typographical errors are corrected for Unit 2 as described in Attachment 1.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has performed an evaluation of the hazards considerations associated with the proposed Technical Specification amendments utlizing the criteria in 10 CFR 50.92. Our evaluation is provided below and specifically addresses the three criteria of 10 CFR 50.92(c) for the changes described above.

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the changes reflect either a more restrictive limit on increases in coolant leakage and a more stringent plant shutdown requirement (Item 1) or administrative changes (Items 2 and 3). The proposed changes do not impact the current operation of plant equipment important to safety nor do they allow any new equipment or new modes of operation.

The proposed amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated because the changes are largely administrative and deal with monitoring requirements, action levels, and surveillances. No new or different modes of operation or changes to plant equipment are allowed by the changes.

The proposed amendments to not involve a significant reduction in a margin of safety because the Item (1) change involves a more restrictive limit on coolant leakage increases, requires plant shutdown if this limit is exceeded regardless of the source of leakage, and establishes reasonable equipment operability requirements consistent with previous NRC guidance. Items 2 and 3 reflect administrative changes which have no functional impact on operation of the plant and therefore, no impact on the margin of safety.

For the reasons stated above, Commonwealth Edison finds that the proposed amendments do not involve a significant hazards consideration based on the criteria of 10 CFR 50.92(c). We, therefore, request approval of the proposed amendment under the provisions of 10 CFR 50.91(a)(4).

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