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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 75 TO PROVISIONAL OPERATING LICENSE NO. DPR-19

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-237

1.0 INTRODUCTION AND DISCUSSION

By letter dated December 21, 1982 (Ref. 1) Commonwealth Edison Company (CECo) (the licensee) proposed Technical Specification changes to allow plant operation at rated conditions following completion of actions planned for the outage and refueling (Cycle 9 reload) which began on January 8, 1983. The core for the Dresden-2 Cycle 9 (D2C9) reload will contain 224 new assemblies. 220 new assemblies are provided by Exxon Nuclear Company (ENC) of which 216 are 8x8s and 4 are lead 9x9 assemblies. The four other new assemblies are provided by General Electric (GE) and are 8x8 prepressurized lead test assemblies. The latter four assemblies were added to monitor the extent of cladding hydrogen uptake due to the injection of hydrogen into the primary coolant for the purpose of reducing Intergranular Stress Corrosion Cracking (IGSCC) in the piping of the primary coolant system.

In a letter dated February 1, 1983, the licensee proposed a technical specification change requesting changes in the set points of three safety/relief valves. This change relates to modifications associated with the Mark I Long Term program.

Other changes in technical specifications have resulted from information obtained during the outage ISI. Non-destructive examination of RCS piping revealed cracks in the recirculation system probably due to IGSCC. Although weld overlay repairs of some cracks are being done, others were judged by the staff to not require repairs till after Cycle 9. However, the staff is requiring changes in the leak detection technical specifications during this period.

Additional information related to the Cycle 9 Reload was submitted by CECo in letters dated February 7 and 24, 1983 and March 10, 11, 18, and 31, 1983. This supplementary information was also used by the staff in its evaluation of the proposed changes to the Operating Technical Specifications for the Dresden Nuclear Power Station, Unit 2.

2.0 EVALUATION OF PROPOSED LICENSE CONDITION AND TECHNICAL SPECIFICATION CHANGES

2.1 FUEL MECHANICAL DESIGN

2.1.1 Background

The Dresden-2 Cycle-9 core will consist of 216 fresh ENC XN-1 8x8 fuel assemblies, 4 ENC 9x9 LTAs, 4 GE P8x8R LTAs, 384 GE 8x8R fuel assemblies, and 116 GE 8x8 fuel assemblies. The ENC XN-1 8x8 fuel design is described in the approved generic report on the jet-pump (JP) BWR fuel design (XN-NF-81-21) (Ref. 19). However, there were four conditions attached to the staff's approval of the use of the ENC JP-BWR fuel design. They are listed below:

(1) The licensee must confirm that the design power profile shown in Fig. 5.10 of XN-NF-81-21 bounds the power limits for the application in question.

(2) Unless RODEX2 (presently under NRC review) is approved without modification, the licensee must confirm or redo the following analyses, which were reviewed on the basis of RODEX2 results: design strain, external corrosion, rod pressure, overheating of fuel pellets, and pellet cladding interaction.

(3) Until such time that XN-NF-82-07 is approved and incorporated in the ENC ECCS evaluation model, a supplemental calculation using the NUREG-0630 cladding models must be provided on a plant-specific basis each time a new ECCS analysis is performed.

(4) The licensee must make sure that the fuel performance code that is used to initialize Chapter 15 accident analyses has current NRC approval.

The staff has evaluated these four conditions during the course of its review, and its conclusions are described in the following paragraphs.

2.1.2 Power History

The licensee stated in the submittal (XN-NF-82-77, Revision 1) (Ref. 1b) that the Dresden-2 Cycle-9 expected power history is bounded by the design power profile in Fig. 5.10 of XN-NF-81-21. ENC also transmitted additional information to confirm the power history. The staff concludes that the Cycle-9 power history is within the design limit, and Condition 1 is satisfied.

2.1.3 RODEX2--Strain, Oxidation, Rod Pressure, Overheating of Fuel Pellets, and PCI Analyses

The analyses of strain, oxidation, rod pressure, overheating of fuel pellets, and PCI were described in the approved JP-BWR fuel design.

However, those analyses were done by using the RODEX2 code, whose review has not yet been completed. Two of the more important of those analyses have been examined in more detail.

(1) Rod Pressure

The Exxon JP-BWR fuel design calls for no rod internal pressure exceeding the system pressure during normal operation. Since Exxon used the RODEX2 code (XN-NF-81-58) (Ref. 13), whose review has not yet been completed, to demonstrate complying with the design basis, the NRC staff requested Exxon to redo the analysis by using the approved GAPEXX code (XN-73-25). An Exxon re-analysis for a burnup up to 12,000 MWd/MTU shows that the internal rod pressure will not exceed the system pressure. Since the approved GAPEXX code includes a correction for the effects of high burnup on fission gas release, the staff concludes that the rod pressure will not exceed system pressure at the end of Cycle-9 operation.

(2) Fuel Centerline Melting

The design basis for Exxon fuel centerline temperature is that no fuel melting should result from normal operation including transient occurrrences. Exxon has generated a fuel temperature history for the Dresden-2 reactor using RODEX2 at a peak rod power of 13.93 kW/ft. Fuel temperatures under this condition envelope maximum temperatures expected at any exposure for XN-1 fuel in Dresden-2. The maximum calculated temperature is 3909°F and occurs at a rod exposure of 21,200 MWd/MTU. By raising the power to 120% overpower (16.72 kW/ft) at this exposure to simulate a limiting transient condition, RODEX2 gives a fuel centerline temperature of 4607°F. The fuel melting temperature at 21,200 MWd/MTU according to an acceptable correlation is 4959°F. Thus, a significant margin to centerline melt exists for 120% overpower.

The staff will require the licensee to confirm or redo the analyses of strain, oxidation, rod pressure, overheating of fuel pellets, and PCI following the approval of the RODEX2 code according to the requirement described in Condition 2. On the basis of the licensee's favorable analyses with a new unapproved code, a limited re-analysis with an older approved code, and this requirement for confirmation when the RODEX2 review is completed, the staff concludes that operation of Cycle 9 is acceptable with respect to strain, oxidation, rod pressure, overheating of fuel pellets, and PCI.

2.1.4 Cladding Swelling and Rupture

The cladding swelling and rupture model in XN-NF-82-07 (Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model) was recently approved for use in the ENC ECCS evaluation model and has been incorporated in the approved ENC EXEM/BWR ECCS model. Since ENC used that approved swelling and rupture model for cladding in ECCS analysis, Condition 3 has been satisfied.

2.1.5 LOCA Initial Conditions

ENC used the approved steady-state code, GAPEXX (XN-73-25), to calculate Cycle-9 LOCA initial conditions including stored energy and rod pressure for the ENC EXEM/BWR evaluation model. Since the version of the GAPEXX code that was used includes a correction for the effects of high burnup on fission gas release, the staff finds this analysis acceptable. Thus Condition 4 is satisfied by the use of the approved steady-state code GAPEXX.

2.1.6 Lead Test Assemblies (LTAs)

There are four ENC 9x9 LTAs and four GE P8x8R LTAs in the Cycle-9 core. The purpose and benefits of testing 9x9 fuel were presented by ENC during a meeting between ENC and the NRC staff (July 14, 1982). The locations of these four LTAs were selected on the basis of allowing adequate exposures and not resulting in additional operating limits. The irradiation of small numbers of LTAs in non-limiting positions does not create a risk to public health and safety and usually provides data that leads to safety improvements.

The installation of four GE LTAs was proposed after the Cycle-9 initial submittal. The purpose of inserting the four fresh P8x8R assemblies, which are of current GE design used in many reload applications, is to monitor cladding hydrogen uptake due to the introduction of a new technique of hydrogen injection into the coolant water (See EPRI Journal, January/February 1983). Hydrogen injection is intended to alleviate the stress corrosion problems in BWRs and prolong the lifetime of pipes and nozzles. However, hydrogen injection could also increase the tendency for cladding embrittlement and thereby increase the probability of cladding failure as a long-term effect on fuel performance.

Because there have been no fuel failures in recent years caused by hydrogen embrittlement, and because that probability is deemed small, the staff concludes that there is reasonable assurance that the new hydrogen injection technique will not cause problems during Cycle-9 and subsequent operations. However, the staff requests that the licensee report the results of any cladding hydrogen uptake measurements related to hydrogen injection and the GE LTAs.

2.1.7 Staff Position

The NRC staff has reviewed the Dresden-2 fuel design and analyses for the Cycle-9 reload. On the basis of supplemental analyses using the approved GAPEXX code and the approval of the generic report, XN-NF-81-21, (Ref. 19) it concludes that the application is acceptable with regard to the mechanical design with the following condition:

Following approval (expected by summer of 1983) of the RODEX2 code, and prior to the end of Cycle-9 operation, the licensee must confirm the adequacy of the present calculations of design strain, external corrosion, rod pressure, fuel centerline temperature, and transient strain for PCI.

2.2 NUCLEAR DESIGN

The Dresden-2 Cycle-9 (D2C9) reload will consist of 774 assemblies of which 224 will be new assemblies that have been described previously. The old assemblies consist of 92 GE 8x8 of 2.5% enrichment, 24 GE 8x8 2.62% enrichment and 384 GE P8x8R of 2.65% enrichment. The 8x8 design consists of 63 fuel rods and one water rod and the 9x9 design consists of 80 fuel rods and one water rod. The average assembly enrichment is 2.83% which includes a six inch natural uranium blanket at both top and bottom. The average enrichment of the central region excluding the blanket is 3.02%.

ENC has calculated K_{∞} for 8x8 and 9x9 reload fuel and for a comparable GE fuel design. Based on the criteria in NEDE-24011, ENC concluded that adequate margin exists for storage of both 8x8 and 9x9 ENC reload fuel in the dry storage vault and the spent fuel pool for the GE designed storage racks. For the high density fuel storage racks designed by Nuclear Services Corp. (NSC), criticality analyses have been performed by NSC for ENC fabricated fuel assuming a center zone enrichment of 3.02% to demonstrate that the $K_{eff} \leq .95$ requirement is met. These NSC calculations are based on bundle reactivity comparisons provided by ENC. These calculations demonstrate that both ENC 8x8 and 9x9 fuel meet the $K_{eff} \leq .95$ requirement, when the gadolinia is taken into account, for the high density fuel storage racks.

The shutdown margin of the new core meets the technical specification requirement that the core be at least 0.25% ΔK subcritical is the worst reactive condition when the highest worth control rod is fully withdrawn and all other rods are fully inserted. For D2C9, ENC calculated that the K_{eff} under cold conditions and the strongest rod out is equal to 0.989 resulting in a shutdown margin of 1.1% ΔK (XN-NF-82-77(P) Revision 1) (Ref. 1b). The effect of settling of B_4C in the absorber tubes of the control rods is not significant for the shutdown margin.

The standby liquid control system is capable of bringing the reactor from full power to a cold shutdown condition assuming one of the withdrawn control rods is inserted. The 600 ppm boron concentration will bring the reactor subcritical to K_{eff} = 0.950 at 70°F xenon free conditions (XN-NF-82-77(P) Revision 1) (Ref. 1b).

Based on its review of the licensee submittal (T. J. Rausch to H. R. Denton, December 21, 1982), the plant specific analysis, XN-NF-82-77(P), Revision 1 and the BWR methodology reports XN-NF-80-19, Volume 1 and Supplements 1 and 2) (Ref. 6), the staff has determined that the nuclear characteristics and the expected performance of the reload core D2C9 are acceptable.

2.3 THERMAL-HYDRAULIC DESIGN

2.3.1 Background

The staff review of the thermal-hydraulic aspects of Dresden 2 Cycle 9 consisted of the following:

(a) the compatability of ENC and GE fuel bundles
(b) the operating safety limit minimum critical power ration (OLMCPR)
(c) thermal-hydraulic stability.

Aspects (a), and (b) were dealt with in detail in the staff review of the Dresden 3 Cycle 8 reload (Ref. 2,3) and this review has concentrated on assuring that these findings are acceptable for Dresden 2 (see Table 1). The thermal-hydraulic stability question was the main focus of the staff review due to the inclusion in the core of the four (9x9) lead test assemblies which have less stable operating characteristics than the 8x8 assemblies. The objective of the review was to confirm that the thermal-hydraulic design of the reload core was accomplished using acceptable analytical methods, provided an acceptable margin of safety from conditions which would lead to fuel damage during normal operation and anticipated operational occurrences (AOOs), and is not susceptible to thermal-hydraulic instability.

2.3.2 Hydraulic Compatability

Since a BWR core is a series of parallel flow channels connected to a common lower and upper plenum, the total pressure drop across the bundles will be equal. However, differences in the hydraulic resistances of the fuel designs may cause variations in axial pressure drop profiles across the bundles.

In response to a staff question during the review of Dresden 3 Cycle 8, the licensee supplied a figure of pressure drop versus axial length for both fuel types. The calculation of these pressure drops was performed using the methodology documented in XN-NF-79-59 (Ref. 4) and hydraulic resistance factors obtained from single-phase flow tests conducted on the fuel bundles. The results of these analyses showed that the differences in local pressure drop is the ΔP across the lower tie plates for the two fuel designs which results in a different total flow for the different assembly types. The ΔPs across the rodded region are of the same magnitude and the total pressure drop equalizes once the upper tie plate is accounted for in the calculations.

Table 1

THERMAL-HYDRAULIC PARAMETER COMPARISON

General Characteristic	Unit	Dresden 3 Cycle 8	Dresden 2 Cycle 9
Core Power Level	(MWt)	2527	2527
Core Inlet Flow Rate	(lb m/hr)	98×10^6	98×10^6
Core Inlet Enthalpy	(BTU/lb m)	522.3	522.9
Steam Dome Pressure	(PSIA)	1020	1020
Upper Plenum Pressure	(PSIA)	1026	1026
Turbine Pressure	(PSIA)	965	965
Feedwater Flow	(M lb/hr)	9.8	9.8
MCPR Limit	1.05(ENC) 1.06 (GE 8x8R)		1.05(ENC) 1.06 (GE 8x8R)
Limiting Transient	Generator Load Rejection (w/o bypass)		Generator Load (w/o bypass)
OLMCPR	1.33 (8x8) 1.34 (8x8R)		1.33 (8x8) 1.35 (8x8R) 1.38 (9x9)

Additional analyses of the effects of hydraulic compatability on thermal margin were presented in the Dresden 3 Cycle 8 reload report. Based on the data supplied for Dresden 3 and on calculations submitted for Dresden Unit 2 showing the hydraulic compatability of the 9x9 and 8x8 assemblies, the staff concludes that the hydraulic compatability between Exxon and GE fuel assemblies is acceptable.

2.3.3 Minimum and Operating Limit CPR

The methodology for determining uncertainties and their application in determining the MCPR limit is contained in XN-NF-80-19 Volume 1 (Ref. 6) and XN-NF-512 (Ref. 15) and XN-NF-524 (Ref. 16). XN-NF-80-19 Volume 1 has been reviewed and approved by the staff.

Although the staff has not completed its generic review of XN-NF-512 and XN-NF-524, the advanced status of its review allows the staff to conclude that the methodology for applying the XN-3 mean and standard deviation to arrive at a 1.05 MCPR for ENC fuel and a 1.06 for GE 8x8R fuel is acceptable.

Various transients could reduce the MCPR below the intended safety limit. The most limiting of these operational transients have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR). Table 2 contains the results of these analyses. The transient which resulted in the greatest ΔCPR was the load rejection without bypass.

The ΔCPR for the load rejection without bypass was calculated using the statistical methodology described in XN-NF-81-22 (Ref. 12). Based on this analysis the licensee has proposed a ΔCPR of 0.26 at a 95% probability level. The addition of this ΔCPR to the safety limit MCPR results in an operating limit MCPR (OLMCPR) of 1.31 for the ENC and GE 8x8 fuel designs, 1.32 for the GE 8x8R design, and a 1.35 for ENC 9x9 fuel.

Until the staff completes its generic review of XN-NF-79-71 (Ref. 5) and XN-NF-81-22 (Ref. 12), the staff will require that code uncertainties be accounted for using the methods discussed in the safety evaluation report on the GE ODYN code (Ref. 18) as described for implementation in its safety evaluation for Dresden Unit 3 (Ref. 2).

Such a procedure for Dresden Unit 2 requires that an ENC code uncertainty value of 0.22 ΔCPR/ICPR, be applied deterministically to CPR calculations. When this ΔCPR is added to the MCPRs the resultant OLMCPRs are 1.34 for ENC and GE 8x8 fuel designs and a 1.35 for GE 8x8R fuel, and 1.38 for ENC 9x9 fuel.

The staff concludes that such an increase in ΔCPR acceptably bounds the code uncertainties and the limits so derived will assure that the safety limit MCPR is not violated in the event of any anticipated transients.

Table 2

THERMAL MARGIN SUMMARY

Transient	ΔCPR/OLMCPR			
	8x8(ENC)	9x9(ENC)	8x8R(GE)	8x8(GE)
Generator Load Rejection (w/o bypass)	.26/1.31	.30/1.35	.26/1.32	.26/1.31
Increase in Feedwater Flow	.21	.26	.21	.21
Loss of Feedwater Heating	.16	.21	.16	.16

2.3.4 Thermal-Hydraulic Stability

Since the major difference between Dresden 3 Cycle 8 and Dresden 2 Cycle 9 is the inclusion of the four 9x9 lead assemblies and since the 9x9 assemblies are less stable than the 8x8 assemblies, a major portion of the staff review of this reload submittal concentrated on assuring that no instability problems would be caused by the lead test assemblies.

This assurance has been provided by two separate means. First, Commonwealth Edison will perform special stability tests and monitoring during Cycle 9 operation. Second, audit calculations of Exxon's decay ratio calculations were performed.

2.3.4.1 Stability Test and Cycle 9 Monitoring Requirements

Commonwealth Edison proposed stability testing and monitoring for Cycle 9 (Ref. 21) based on a meeting held with the staff at the Dresden site on February 10, 1983. Their program, which is described below, is acceptable to the staff.

Stability Test

A. Initial Conditions

1. The test will be performed after preconditioning the reactor core for full power operation.
2. The test will be performed at the intersection of the minimum recirculation pump speed (two pump operation) and approximately the 100% power/flow line ($\sim$38% of rated flow and $\sim$54% of rated power).
3. The control rod pattern will correspond to the nominal full power pattern.

B. Data Acquisition

1. Prior to test initiation, baseline data will be recorded to define the initial conditions for the test. This will include reactor power, core flow, core pressure, control rod pattern, LPRM readings, cycle exposure, and nuclear limits on power distribution (from the POWERPLEX Core Monitoring System). This data will be taken at the reactor conditions specified in Section A.2 above.

2. During the test, local and core-wide power response will be monitored with a multi-channel strip chart recorder. The following instrument response will be recorded:

   a. 1 or 2 APRM channels
   b. The B, C and D level LPRMs from LPRM string 16-17
   c. The B, C and D level LPRMs from LPRM string 16-09 or 08-17

The exact LPRMs and APRM(s) to be recorded will be determined prior to the test considering LPRM and APRM operability/availability. The nominal chart speed will be 1 inch/second to provide resolution of the expected power oscillation.

C. Test Criteria for Initiating Corrective Action

1. If the LPRM signals being monitored exhibit divergent oscillations, the actions specified in section E below will be taken. An oscillation will be considered divergent if its amplitude increases by 5 or more watts/cm^2* relative to the initial flux peak as observed on the strip chart recorder or the Rod Block Monitor LPRM display.

2. Prior to performing the test, LPRM action levels will be determined for each LPRM being monitored as follows:

$$\text{LPRM Action Level} = \frac{0.95\ (\text{initial LPRM Reading})\ (\text{initial CPR for 99 assembly})}{(\text{full power CPR operating limit for 9x9 assembly})}$$

where the initial values correspond to values obtained prior to test initiation as identified in section B.1 above (with control rod D-4 at the normal, full power position).

If sustained LPRM oscillations are observed (decay ratio = 1) such that their amplitude exceeds the above defined criteria, the actions specified in section E below will be taken.

3. If sustained LPRM oscillations are observed with an amplitude that does not exceed the action levels from C.2 above, withdrawal of control rod D-4 will be terminated although data acquisition may continue. Upon completion of data acquisition (as determined by the cognizant engineer), the actions specified in section E below will be taken.

4. If divergent or sustained APRM oscillations in excess of 15% peak-to-peak are observed on the strip chart recorder or the normal APRM chart recorders, the actions specified in section E will be taken.

D. Test Initiation

The local reactivity perturbation shall be accomplished by full insertion of Control Rod D-4 followed shortly thereafter by continuous withdrawal of Control Rod D-4 to its initial, pre-test position.

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*Although the LPRMs are calibrated in heat flux units for steady-state operation, they are responding proportional to the neutron flux under these transient conditions.

This control rod is selected due to its close proximity to the monitored 9x9 fuel assembly.

E. Immediate Actions

1. Terminate withdrawal of control rod D-4.
2. Insert control rod D-4 to position 00.
3. Insert additional control rods as necessary to terminate power oscillations (specific control rods will be identified prior to test initiation).

F. Subsequent Actions for Continued Operation

1. In the event that no sustained or divergent power oscillations are observed during the test, normal ascension to power may continue upon completion of test.

2. In the event that sustained or divergent power oscillations are observed such that the section E actions are invoked, normal operation may continue according to the following:

   a. Verify that power oscillations have been damped (returned to pre-test noise levels).
   b. Insert control rods as necessary to reduce the flow control line by 5%.
   c. Increase core flow by 5%.
   d. Perform single notch withdrawal of control rod D-4 to its position prior to stability test initiation (normal full power position).
   e. Proceed with normal power ascension using flow control.
   f. Additional control rod withdrawal is allowed to increase the flow control line providing that core flow is at least 60% of rated flow.
   g. Refer to Section G.

G. Long Term Corrective Actions (duration of Cycle 9)

In the event that either (a) sustained or divergent oscillations are observed such that the section E actions are invoked or (b) the oscillatory behavior of LPRM location 16-09 (adjacent to the 9x9 assembly) is inconsistent with the behavior of the monitored LPRMs adjacent to an 8x8 assembly, the need for restrictions on future operation during Cycle 9 will be determined by the Dresden On-Site Review Committee. Recommendations will be formulated as necessary in consultation with CECo's Nuclear Fuel Services Department, Off-Site Review, Exxon Nuclear and the NRC, based on the oscillations observed during the test. The need for supplemental LPRM monitoring during Cycle 9 will also be determined.

Proposed Monitoring Requirements for D2C9

1. Existing hard-wired LPRM alarms at Dresden Station provide continuous monitoring for abnormal LPRM indications. If excessive local power oscillations occur during Cycle 9, the alarm setpoint will be exceeded resulting in an audible and visual alarm in the control room. Station procedures will require that the operator insert control rods to suppress local oscillations if sustained or oscillatory alarm indications are observed. A separate alarm is provided for each LPRM detector. LPRM alarm setpoints will be established to ensure that power oscillations would be detected prior to achieving levels that would correspond to the MCPR Safety Limit.

   Since unusual local power oscillations are not expected during the local stability test or during normal operation, the above described LPRM monitoring will provide adequate protection for Cycle 9 operation. Substantial BWR operating experience at Commonwealth Edison and throughout the industry has demonstrated the strong neutronic coupling of BWR cores. The occurrence of significant local power oscillations will result in core-wide power oscillations readily observable on the APRMs which provide automatic alarm, rod block and reactor scram functions.

   If unusual power oscillations are observed during the stability test, the need for additional LPRM monitoring during Cycle 9 will be re-evaluated by the Dresden On-Site Review Committee (see Test section G).

2.3.4.2 Audit Calculations

In order to assure margin to instability for the Dresden 2 Cycle 9 core exists, bounding audit calculations were performed on an equilibrium 8x8 and equilibrium 9x9 core. These calculations are bounding for two reasons. First, cores tend to become more unstable at higher burnups and these calculations were performed on equilibrium cycle high burnup fuels with more burnup than the current cycle fuel. Second, an all 9x9 core results in a more unstable configuration than a core with only four 9x9 assemblies.

Based on the calculations presented in Table 3, the staff concludes that:

1. Reasonable agreement exists between the stability calculations performed at ORNL using LAPUR and Exxon's stability calculations.
2. Calculations indicate stability margin exists for the all 9x9 core in the Dresden plant, and the results of the all 9x9 core bound the conditions of the (9x9) lead test assemblies for Dresden Unit 2 Cycle 9.

Table 3

DECAY RATIOS CALCULATED AT NATURAL CIRCULATION CONDITION

(32% RATED FLOW-47% RATED POWER) FOR EQUILIBRIUM BURNUP FUEL

	EXXON*	ORNL**
FULL CORE 9x9 ASSEMBLIES	.73	.80
FULL CORE 8x8 ASSEMBLIES	.63	.63

* Using COTRAN CODE

** Using LAPUR Code with same input cross sections as the COTRAN code.

3. Since the staff has determined that there is considerable uncertainty in the calculated results, it does not find that these results alone justify the loading of a full core of 9x9 assemblies. The results of the Commonwealth Edison testing program during this cycle and further review in the NRC generic assessment of thermal-hydraulic stability will be required before a judgement on a 9x9 fuel loading can be made.

2.4 TRANSIENTS AND ACCIDENTS

The NRC staff has reviewed the analyses of Fuel Misloading, Control Rod Withdrawal Error, Rod Drop and LOCA and its evaluation follows:

2.4.1 Fuel Misloading

Fuel misloading and fuel misorientation analyses were performed using the methodology described in XN-NF-80-19, Volume 1 and Supplements 1 and 2 (Ref. 6). The analyses covered both types of GE fuel and the ENC 8x8 fuel. The largest ΔCPR calculated was 0.14 for a misloading or misorientation error. The misloading error is bounded by the load rejection without bypass transient which constitutes the limiting transient for D2C9. (XN-NF-82-77(P), Revision 1 and XN-NF-82-84(P), Revision 1) (Refs. 1b and 1c).

2.4.2 Control Rod Withdrawal Error

The control rod withdrawal error, which accounts for the inadvertent withdrawal of a high worth control rod, was calculated using the methodology described in XN-NF-80-19(P), Volume 1 and Supplements 1 and 2. The rod block setting was increased from 107% to 110% and the resulting ΔCPR was 0.13 which is less than the limiting value of 0.26 calculated for the load rejection without bypass transient. (XN-NF-82-77(P), Revision 1).

2.4.3 Rod Drop Accident

A control rod drop accident analysis was performed using the parametric values developed in XN-NF-80-19, Volume 1 and Supplements 1 and 2. The calculated enthalpy deposition was 111 cal/gm, which is far less than the allowable limit of 280 cal/gm. In this analysis the dropped control rod worth was 0.78% Δk, the Doppler coefficient was -9.8×10^{-6}/°F, the effective delayed neutron fraction was 0.0055 and the four assembly local peaking factor was 1.19.

2.4.4 LOCA

Maximum Average Planar Linear Heat Generation Rates were calculated by Exxon from a LOCA analysis performed for the Dresden 2 reactor. These results were obtained using the NRC approved EXEM (Ref. 20). The generic jet pump BWR 3 LOCA break spectrum analysis (Ref. 14) shows the limiting break for a BWR 3 on a generic basis to be double-ended guillotine configuration in the recirculation piping on the suction side of the pump. The MAPLHGR heatup analysis for Dresden Unit 2 Cycle 9 utilized this limiting break.

Identification of the Dresden 2 Design Basis LOCA as the limiting transient on the basis of the generic BWR 3 break location and spectrum studies previously reviewed and approved is considered acceptable, particularly in view of the margin (~300°F) to the limit value.

Based on its review of XN-NF-82-88, "Dresden Unit 2 LOCA Analysis Using EXEM/BWR Evaluation Model MAPLHGR Results" (Ref. 1d), the staff finds that the calculations are performed according to the requirements of 10 CFR 50 Appendix K and that the MAPLHGR limits presented in this report satisfy the ECCS criterion specified by 10 CFR 50.46 and are acceptable.

Based on the discussion presented above, the staff concludes that with respect to the transients and accidents described, operation of Dresden 2 Cycle 9 is acceptable.

2.5 SUMMARY

Based on its review, the staff concludes that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses for Dresden 2 Cycle 9 are acceptable with the following conditions:

1. Presently the staff is reviewing the uncertainties associated with the ENC plant transient code on a generic basis in conjunction with our evaluation of XN-NF-79-71 (Ref. 5) and XN-NF-81-22 (Ref. 12). Until the review of this topic is complete, the staff has required that a code uncertainty of 0.022 be deterministically applied using the method discussed in the ODYN SER. This method should appropriately bound the expected result from the staff's generic review. When code uncertainties are considered, the result is an increase in ΔCPR from the reported value of 0.26 to a new value of 0.29. Adding this ΔCPR to the MCPR yields an operating limit MCPR of 1.34 for the ENC and GE 8x8 fuel, 1.35 for the GE 8x8R fuel design, and 1.38 for ENC 9x9 fuel.

2. Commonwealth Edison will perform the stability test and monitoring as described in Section 2.3.4.1 of this Safety Evaluation.

3. Following approval (expected in the next few months) of the RODEX2 code and prior to the end of Cycle 9 operation, the licensee must confirm the adequacy of the present calculations of design strain, external corrosion, rod pressure, fuel centerline temperature, and transient strain for PCI unless RODEX2 is approved without modifications.

2.6 SAFETY RELIEF VALVE SETPOINT CHANGES

In analyses associated with the Mark I containment program, it was discovered that the torus could be subjected to excessive loads if a relief valve actuation occurs shortly after closure. This loading is the result of a water leg entrapped in the relief valve discharge line from the vacuum caused by the condensed steam in this line. To prevent such loading, a modification to the electromatic relief (EMR) valve logic is currently being installed which will delay automatic opening of two EMR valves up to ten seconds from the last closure of the valve. In order to maintain very similar overall Target Rock and EMR valve performance with the logic change and prevent excessive loading, the two affected EMR valves' (203-3B and 203-3C) TS pressure setpoints must be lowered so that they are the first to actuate and the setpoint of one valve (Target Rock) will be raised.

Calculations have shown that the containment isolation pressurization event is the most limiting of the pressurization transients (Ref. 1b). Calculation of this event, which results in closure of all steam isolation valves without direct scram and relief valve operation, was computed with the COTRANSA version of ENC's Plant Transient Simulator. This version was used to provide adequate consideration of the core void collapse produced by pressure wave propagation into the reactor vessel before reactor scram is actuated. Conservative assumptions concerning the neutron flux trip level, control rod insertion time, control rod worth, void reactivity feedback, and end of core life conditions for fuel gap conductance were used to provide a conservative bounding system pressure transient. Results of the analyses demonstrated a peak pressure in the lower plenum of 1364 psia or 26 psi below the ASME Pressure Vessel Code Limit of 1390 psia for the Dresden 2 vessel (1.10 x Design 2 = 1.10 x 1250 psig = 1375 psig). The corresponding steam dome pressure is 1325 psig. The TS safety limit is based on dome pressure. The staff, therefore, concludes that the licensee's proposed RCS safety limit of 1345 psig is conservative with respect to the above evaluation and is therefore acceptable.

The delay in actuation between successive valve openings is required because of the possibility of an automatic depressurization immediately following opening of a valve for pressure relief. The calculated minimum acceptable delay time reported by the licensee is 6.2 seconds. This compares conservatively with the ten-second delay proposed for the TS, with ample margin.

To avoid the possibility of premature actuation of the two EMR valves, the actuation logic has been modified by adding contacts from "time delay on de-energization" relays in the opening circuitry for each valve. These contacts are normally closed. Upon EMR valve actuation, these contacts open and remain open for ten seconds after the valve has closed, thus blocking all automatic EMR valve actuation signals during this interval.

After the ten seconds have elapsed, the contacts re-close to allow subsequent EMR valve actuations if required. During the ten second interval when EMR valve automatic actuation is blocked, manual capability to open the valves via hand switches on the main control board still exists. An amber light indication at each control switch is provided during this ten second interval. The plant operating procedures have been revised to instruct the operators not to open either of these two valves when the associated amber light is lit.

Both valves 203-3B and 203-3C are automatic depressurization system (ADS) valves. Power to the actuation circuits for each valve may be provided by one of two separate sources. Normally power is provided by 125 Vdc main bus No. 2. If this supply should fail, power to the circuits will be automatically transferred to 125 Vdc reserve bus No. 2. Each of these is a Class 1E safety bus. The staff discourages the use of automatic transfer schemes in new designs since the potential exists for electrical faults to challenge redundant safety circuits. However, since this automatic transfer feature is part of the existing protection system (ADS) circuitry at Dresden Unit 2, and, since the licensee has stated that redundant protective devices (fuses and circuit breakers) are located between the automatic transfer and the safety buses and that these protective devices are adequately sized to clear any faults, the staff finds the design to be acceptable. Annunciation is provided in the control room upon loss of power to the EMR valve actuation circuits.

The licensee has indicated that the added relays are safety grade. The added circuitry (for both valves) is located in an instrument panel in the auxiliary electrical room which has a controlled environment. The only bypass capability for this EMR valve actuation circuitry is via the three position (MANUAL-OFF-AUTO) hand control switches for the two valves when in the "OFF" position. Valve position status lights at the control switches are extinguished when a switch is placed in the "OFF" position, but no annunciation is provided. The Dresden 2 Technical Specifications do not allow plant operation with an ADS valve out of service for more than seven days. However, if one of the two valves is bypassed, then a single failure in the circuitry for the remaining valve could prevent these valves from performing their function (i.e., preventing excessive torus loads due to successive EMR valve openings). It is the staff's conclusion that this capability for bypass is acceptable since these valves are rarely bypassed, there is a low probability of needing this function during the short period of time when the bypass is in effect, and there is the capability of the torus to handle discharge loads several times during the life of the plant.

The licensee has indicated that the added EMR valve actuation circuitry will be fully tested upon installation and at each refueling interval thereafter. This testing will include the time delay setting and the operation of the amber lights on the main control board and is consistent with the required Technical Specification surveillance interval for testing the remaining ADS functional logic. A commitment to perform this testing has been documented by the licensee.

The staff also notes that there are single failure points within this design which, if one occurs during the ten second interval, could cause one of the EMR valves to open. However, the staff considers the probability of such a failure occuring during this interval to be sufficiently low so as not to be a concern. The above design will prevent single failures that could occur within the modified circuitry from preventing valve actuations after ten seconds, thereby, preventing multiple simultaneous re-openings of the remaining safety-relief valves.

Based on this review, the staff has determined that the EMR valve actuation circuitry modifications described above are designed to perform their intended function given a single failure. The circuit modifications proposed by the licensee are designed in accordance with the requirements of IEEE Standard 279, and, therefore, are acceptable.

The staff has reviewed the proposed changes in the SRV setpoints, the proposed delay for successive actuations and the logic circuits designed to prevent premature reactuation of the SRVs and finds the changes to have minimal effect on safety limits and, therefore, to be acceptable.

2.7 INSPECTION AND REPAIR OF THE DRESDEN 2 RECIRCULATION SYSTEM PIPING

2.7.1 Introduction

During the current 1983 refueling outage, augmented inservice inspection was performed on 47 austenitic stainless steel welds in the 12", 22", and 28" recirculation system piping and on 2 austenitic stainless steel welds in each of the 16" Low Pressure Coolant Injection System (LPCI) and shutdown cooling system piping. A total of fifty-one welds were inspected, which include all furnace sensitized safe ends of 28" and 12" diameters. The number of welds and percentage of welds inspected for each size of piping are listed in Table 4. The results of ultrasonic tests (UT) indicated that one 28" recirculation safe end (furnace sensitized) to elbow weld in loop B and nine 12" riser to elbow welds showed reportable linear indications.

The original sampling size consists of only 23 welds, but was subsequently expanded to 51 welds when linear indications were reported on the furnace sensitized safe end weld and riser elbow welds. With

Table 4

AUGMENTED INSERVICE INSPECTION

Dresden Unit 2
March, 1983

System/Category	Diameter	Butt Welds Examined	Butt Welds Total	Reportable Indications	% of Welds Inspected
Furnace Sensitized SE	28"	2	2	1	100%
	12"	10	10	None	100%
Recirc Outlet	28"	6[1]	33	None	18%
Recirc Ring Header	22"	4	10	None	33%
	Sweepolet	2[2]	8	None	
LPCI	16"	2	6[3]	None	33%
Shutdown Cooling	16"	2	6[3]	None	33%
Recirc Inlet Piping	12"	14[4]	40[4]	2	
1st Additional Sample	12"	4	26	1	40%
2nd Additional Sample	12"	17	22	6	

1 Includes two piping to safe end welds
2 Adjacent to end caps
3 Up to isolation valve
4 Includes ten piping to safe end welds

a few exceptions, the welds were selected for inspection in each piping line primarily based on the stress rule index (SRI) considerations. Lambert, McGill and Thomas, a NDE contractor and Commonwealth Edison (CECo) UT personnel performed the UT inspection. Their UT procedures and calibration standards were satisfactorily evaluated on Intergranular Stress Corrosion Cracking (IGSCC) cracked pipe samples at Battelle-Columbus in accordance with I&E Bulletin 82-03.

General Electric Company performed the evaluation of the flaws found in the ten welds mentioned above. NUTECH prepared the weld overlay repair program for seven flawed riser elbow welds. The overlay has a minimum thickness of 0.2 inch and a minimum length of 4.5 inches. GE's analysis has shown that the three unrepaired welds (one recirculation safe end to elbow and two riser elbow welds, PD5-D20 and PD2-D5) will continue to have the original design safety margin for at least one fuel cycle.

In a response to the NRC staff's concern regarding the uncertainties in predicting the crack growth in pipes, CECo proposed to implement the following to ensure that any excessive crack growth in unrepaired welds will be identified in mid-cycle and that early corrective action will be taken for excessive unidentified leakage:

(1) Mid-cycle (9 ± 3 months) UT inspection on the three flawed and unrepaired welds (one 28" recirculation safe end to elbow weld and two 12" riser to elbow welds) will be performed.
(2) The additional leak detection requirements listed in Section 2.7.6 (Leak Detection) will be implemented.

2.7.2 Description of Cracks

One circumferential indication estimated to be approximately one inch long and 0.20 inch deep (16% of the 1.22 inch wall) was found on the safe end to elbow weld of the 28 inch recirculation outlet nozzle (NIB). In addition, several small spot indications no more than 5% of through wall thickness in depth were also detected. All indications are located on the safe end side of the weld. The safe end was furnace sensitized.

Numerous axial and circumferential indications were found on nine stainless steel elbow welds in the 12 inch recirculation risers. All indications are located in the heat affected zones (HAZ) of the elbow welds. The location, orientation and size of each indication are listed in Table 5. The length of the cracks varies from 3/16 inch to 1 1/2 inch and the depth of the cracks varies from 10% to 30% of through wall thickness.

2.7.3 Fracture Analysis

The General Electric Company (GE) has performed an analysis of the flaws for the licensee. The methodology used in the analysis was based on the proposed Appendix X to the new paragraph IWB-3640 of the ASME Code

Table 5

Dresden 2 12" Recirculation Riser
Elbow Welds - Crack Indications and Calculated
Crack Depth After 18 Months

Nozzle/ Line	Weld	Flaw Orientation	Initial Size Length X % Thru wall (0-58in)	Calculated Final depth as % thru-wall @18 months	Overlay Repair
N2A/ 2-0201H -12"	PD4-D23 (Bottom)	Axial	1/2" x 27%	73%	Yes
		Axial	3/8" x 14%	50%	
		Axial	5/8" x 26%	87%	
		Circum	1/2" x 19%	48%	
		Circum	7/16" x 13%	43%	
N2B/ 2-0201J -12"	PD5-D20 (Top)	Circum	1/4" x 17%	38%	No
		Circum	1/4" x 19%	36%	
N2E/ 2-0201M -12"	PD19-D13 (Top)	Axial	3/4" x 19%	95%	Yes
		Axial	1 1/8" x 10%	80%	
	PD19-D14 (Bottom)	Circum	1 1/2" x 24%	81%	Yes
N2F/ 2-0201C -12"	PD7-D11 (Top)	Circum	1 1/2" x 14%	79%	Yes
		Axial	9/16" x 16%	65%	
N2H/ 02-0201E -12"	PD9-D8 (Bottom)	Circum	3/4" x 30%	61%	Yes
N2J/ 2-0201F -12"	PD2-D4 (Top)	Axial	1" x 27%	100%	Yes
		Axial	1/2" x 19%	65%	
	PD2-D5 (Bottom)	Circum	1/2" x 19%	48%	No
		Circum	1/4" x 14%	35%	
N2K/ 2-0201G -12"	PD3-D2 (Bottom)	Axial	1/4" x 23%	44%	Yes
		Axial	1/2" x 14%	58%	
		Axial	1/4" x 14%	38%	
		Axial	3/16" x 21%	39%	

Section XI. This proposed paragraph IWB-3640 and Appendix X have been approved by the ASME Code main committee in March 1983 (Ref. 22) and are expected to be incorporated into the code soon. The determination of allowable flaw size in this methodology is based on the net section collapse approach. GE calculated the final crack size for each flaw at the end of the 18 month fuel cycle and compared it with the calculated allowable flaw size at that location to determine whether a weld overlay repair was needed or not. The calculated allowable flaw size maintains a code safety margin of 2.7 to 3. The results of GE's analysis indicated that the flawed 28" recirculation outlet safe end to elbow weld and 2 of the 9 flawed 12" recirculation riser elbow welds (PD5-D20 and PD2-D5) do not require weld overlay repair, because the calculations show that the existing cracks in those welds will not grow beyond the proposed code allowable crack size during the next 18 month fuel cycle.

2.7.4 28" Recirculation Safe End to Elbow Weld Evaluation

GE has calculated the allowable flaw size for the flawed safe end to elbow weld of the 28" recirculation outlet nozzle in loop B based on the methodology in the proposed Appendix X to IWB-3640 of the ASME Code Section XI. The allowable flaw size is calculated to be 75% of the wall thickness. (The 75% wall thickness is a cut off value to prevent potential leakage from the crack.)

GE has performed an evaluation of the growth of the largest crack found in this weld. This was a circumferential crack, approximately 1 inch in length and 16% of wall thickness in depth, caused by IGSCC. Crack growth due to fatigue is not significant because the cyclic loads caused by startups and shutdowns are small.

In calculating the crack growth due to IGSCC, GE used the computer program G-CRACK (Ref. 23) to calculate the stress intensity factor "K" as a function of crack depth for the uniform sustained axial stresses and the axial residual stresses. The total sustained axial stresses are 11 Ksi, consisting of pressure (6 Ksi), weight (1 Ksi) and thermal (3 Ksi), which are derived from the piping and safe end stress reports, and 1 Ksi of pressure loading on the crack surface. The axial residual stresses are characterized by a 30 Ksi tensile residual stress on the inside diameter surface, which is assumed to be reduced and converted to compressive residual stress at a depth of about 17% of wall thickness. The maximum compressive residual stress of 21 Ksi is assumed at a depth of 40% of wall thickness. GE indicated that this distribution of residual stresses (assumed in this analysis) is typical of large diameter pipes.

The upper bound of GE's constant load crack growth data for furnace sensitized specimens in 8 ppm O_2 environment (Ref. 24) was used in the crack growth evaluation because the safe end was furnace sensitized.

Based on the analysis described above, the final crack depth at the end of an 18 month fuel cycle was calculated to be approximately 44% of wall thickness, which is well within the proposed code allowable crack size of 75% of wall thickness (for relatively short cracks). Therefore, GE concluded that the operation of Dresden 2 with the flawed 28" recirculation safe end to elbow weld is justified for 18 months.

The staff has reviewed GE's analysis and has some reservations regarding the laboratory-developed crack growth rate data used in the calculation. This is because the laboratory test environment may not truly represent the water chemistry surrounding the crack tip and, consequently, the test results may be non-conservative. The staff also has reservations regarding the method used to calculate the crack growth in the presence of a through wall gradient of residual stresses. In particular, the staff is still evaluating whether the distribution of the residual stresses assumed in the analysis and the super-position method GE used in calculating the stress intensity factor for an advancing crack through a complex stress gradient is conservative or not. In spite of the above uncertainties, the staff concludes that the operation of Dresden 2 for one fuel cycle of 18 months with the subject safe end to elbow weld in the as flawed condition is acceptable and does not represent a safety concern. The bases for this conclusion are:

(1) The licensee will ultrasonically inspect the subject safe end to elbow weld and two other flawed riser elbow welds at mid-cycle (9 ± 3 months) to assure there is no excessive crack growth.
(2) Dresden 2 will be operated under hydrogen water chemistry (See Section 2.8), which is expected to reduce the IGSCC crack growth rate.
(3) By inspecting the flaw diagram in GE's analysis, it can be seen that the allowable flaw size will not be exceeded even if the initial crack depth or length is doubled. This should relieve some of the concerns regarding the uncertainties in sizing the circumferential cracks.
(4) As will be discussed later, the licensee will implement augmented reactor coolant leakage detection requirements, which include more frequent monitoring and more restrictive leakage limits.

2.7.5 12" Recirculation Riser Elbow Welds and Repair Evaluation

GE has performed an analysis for each crack found on the nine flawed riser elbow welds. The methodology used in this evaluation is essentially the same as that used in the safe end to elbow weld evaluation. The calculated allowable flaw size based on the proposed Appendix X is 75% of the wall thickness for each flawed riser elbow weld.

In calculating the crack growth, only IGSCC is considered because crack growth due to fatigue is not significant in riser elbow welds. In GE's analysis, a uniform tensile residual stress of 20 Ksi is assumed for axial cracks and a linear through wall distribution of ±30 Ksi residual stresses is assumed for circumferential cracks. GE's crack growth rate for weld sensitized stainless steel in a 0.2 ppm O_2 environment (Ref. 24) was used in the crack growth evaluation. The calculated final crack depth at the end of the 18 month fuel cycle for each crack is listed in Table 5. The results show that six of the nine flawed riser elbow welds would have crack depths exceeding the proposed code allowable flaw size of 75% of wall thickness at the end of the 18 month fuel cycle. The licensee indicated that a total of seven flawed riser elbow welds, consisting of the six elbow welds mentioned above and weld PD2-D4, will be weld overlay repaired to ensure the structural integrity of the subject welds. The riser elbow weld PD2-D4 is conservatively included in the overlay repair program although the calculated final crack size of 61% of wall thickness at the end of the cycle is still within the proposed code allowable crack size of 75% of wall thickness. The licensee indicated that no weld overlay repair will be applied to riser elbow welds PD5-D20 and PD2-D5 because the calculated final flaw sizes of 38% and 48%, respectively, at the end of the 18 month fuel cycle are well within the proposed code allowable flaw size. The licensee concluded that the operation of Dresden 2 with the two flawed riser elbow welds in as-is configuration is justified for 18 months.

The staff has reviewed GE's analysis and has concluded that the operation of Dresden 2 for one fuel cycle of 18 months with the two flawed riser elbow welds (PD5-D20 and PD2-D5) in the as-is configuration is acceptable and does not represent a safety concern. However, the staff has reservations, similar to those discussed in Section 2.7.4, regarding GE's method of analysis. The bases for the staff's conclusions are also similar.

2.7.6 Leak Detection

Although the conservative calculations discussed above indicate that the cracks in the unreinforced welds will not progress to the point of leakage during the next fuel cycle, and very wide margins are expected to be maintained over crack growth to the extent of compromising safety, uncertainties in crack sizing and growth rate still remain. Further, not all welds were examined, and significant cracks could be present in welds that were not examined.

Because of these uncertainties, the staff concluded that the requirements for monitoring for unidentified leakage should be improved.

Commonwealth Edison has agreed to additional monitoring and tighter limits on unidentified leakage. These commitments are summarized below:

(1) Floor drain leakage shall be measured once every 4 hours when the reactor is at operating pressure.
(2) If unidentified floor drain leakage increases by 1 GPM during any 4 hour period, or equals 3 GPM total, action will be taken to identify the source of the leakage.
(3) If unidentified floor drain leakage increases to 4 GPM, a containment entry will be made to determine the source of the leakage.

To ensure that the NRC is aware of any possible problem, Commonwealth Edison will notify the NRC within 24 hours if action points 2 and 3 are exceeded.

If leakage is identifed as coming from a cracked pipe, the plant will be shut down for further investigation and repair.

These additional leakage limits will apply during the next fuel cycle, and will be re-evaluated at the next refueling outage.

2.7.7 Summary and Conclusion

The staff has reviewed Commonwealth Edison Company's submittals dated December 1, 1982, March 1 and March 18, 1983 regarding the actions taken during this refueling outage on the analyses and repairs of recirculation piping system welds in the Dresden Unit 2 plant. This includes description of the defects found, description of repairs to be performed, stress and fracture analysis.

The staff concludes that Dresden Unit 2 can be safely returned to power and operate in its present configuration at least until the next refueling outage, provided the following items are satisfactorily completed:

(1) The Code-required hydrostatic test and nondestructive examination on overlay repaired welds should be satisfactorily completed prior to startup.
(2) The additional leak detection requirements as listed in the section on Leak Detection should be properly implemented prior to startup.
(3) The mid-cycle (9 ± 3 months) UT inspection on the flawed recirculation safe end to elbow weld and two flawed and unrepaired riser to elbow welds (PD5-D20 and PD2-D5) should be satisfactorily performed during the next fuel cycle.

Nevertheless, the staff still has concerns regarding the long term growth of small IGSCC cracks that may be present but not detected during this inspection. Therefore, the staff is also requiring that plans for inspection and/or modification of the recirculation and other RCPB piping systems during the next refueling outage be submitted for staff review before the start of the next refueling outage.

2.8 FULL-TIME OPERATION UTILIZING HYDROGEN ADDITION

2.8.1 Background

By letter dated February 7, 1983, Commonwealth Edison Company (CECo) requested a technical specification change that will allow CECo to increase the Main Steam Line Radiation Monitor trip level setpoints during operation above 20% power to facilitate full time operation utilizing hydrogen addition.

Full time operation of Dresden 2 with 50-70 standard cfm hydrogen injection into the feedwater system is the next phase in the Alternate Water Chemistry program being conducted to reduce Intergranular Stress Corrosion Cracking (IGSCC) of welded type 304 SS in BWR reactor water chemistry. Hydrogen addition to the feedwater suppresses the radiolytic production of oxygen in the reactor coolant which should decrease the susceptibility of austenitic stainless steel to IGSCC.

With standard BWR water chemistry, the bulk of the N^{16} formed from the O^{16} (n,p) reaction is quickly converted to relatively non-volatile anionic species, primarily NO_2^- or NO_3^- . Only a small amount of the N^{16} goes into the steam. As the oxidizing potential of the coolant is reduced by oxygen suppression, since hydrogen additions suppress radiolytic oxygen formation, the proportion of the N^{16} converted to more volatile species such as NH_3, N_2, or NO_2 markedly increases and thus the fraction of N^{16} released to the steam rises commensurately. Consequently, higher radiation levels from N^{16} concentrations in the steam will occur in the main steam line when hydrogen is added to the feedwater. This requires a plant Technical Specification change.

During the 1982 Dresden 2 test when hydrogen was added at the same rate proposed for the long term test, the radiation levels indicated by the main steam line monitor increased by a factor of 4.8 at full power. The results of the 1982 Dresden test are reported in NEDC-23856-7, "Oxygen Suppression in Boiling Water Reactors - Phase 2, Final Report," dated October 1982. In these tests, the main steam line monitor maximum increase was from 458 to 2200 counts/sec or 1.45 to 6.95 R/hr.

2.8.2 Evaluation

In the February 7, 1983 letter, the licensee provided information, including NEDC-23856-7, relating to the full time operation of Dresden 2 with hydrogen injection. The staff used this information to evaluate the following: (1) the technical specification changes that would allow the licensee to increase the Main Steam Line High Radiation Monitor set point; (2) the hydrogen addition system; and (3) the radiological impact.

During operation above 20% thermal power with hydrogen injection, it will be necessary to increase the Main Steam Line High Radiation Monitor scram and isolation trip settings because of the increased N^{16} levels. This radiation monitor's only safety function is in Rod Drop Accident mitigation. However, the Rod Drop Accident is only of concern below 20% thermal power. The capability to monitor for fuel element failures, which could result in increased occupational doses, is maintained through: the continued capability of the radiation monitor to detect failures at $<$20% power; the performance of daily primary water analyses and the trends of these analyses; routine surveys and the capability of downstream process monitors, such as the steam jet air ejector, to detect radioactivity from fuel failure. Dose rate data taken during the one month test in 1982 demonstrated that the increased main steam radiation levels could be readily accommodated by limiting access to certain turbine building areas and that shine at the site boundary is not a concern. Therefore, increasing the Main Steam Line High Radiation Monitor scram and isolation trip settings above 20% power during long-term hydrogen injection is acceptable.

The licensee's hydrogen addition system is designed to reduce the potential hazard to safety-related systems. Central storage of hydrogen is located outside of the plant buildings. Where hydrogen piping is routed through safety-related areas, excess flow valves are provided to prevent the accumulation of a hydrogen concentration greater than 2% in the event of a line break. In addition, all areas containing hydrogen piping are provided with hydrogen detectors which will alarm and isolate the addition system if hydrogen is detected.

Based on the above, the staff concludes that the licensee's hydrogen addition system meets the guidelines contained in Section C.5.d of BTP CMEB 9.5-1 (NUREG-0800), and is, therefore, acceptable.

The staff has evaluated those radiation protection measures implemented and planned by the licensee during the hydrogen injection program to protect the workers and the general public, as well as those actions incorporated to insure that doses incurred are as low as is reasonably achievable. The licensee's actions are consistent with staff guidance in Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," and meet staff requirements in 10 CFR Part 20.1(c) and are therefore acceptable for the technical specification changes proposed to enable the continuation of the hydrogen injection program. The basis for staff acceptance of the changes is the successful conduct of the test program as outlined in NEDC-23856-7. The major ALARA feature is the potential for avoiding doses incurred from pipe cracking repair and non-destructive analysis. Modest operational dose rate and dose increases are offset with operational ALARA measures. By precluding the pipe cracking phenomena, there is a potential for dose savings of hundreds of rem at Dresden 2, and a potential for saving thousands of rem among all BWRs over plant life. Among the ALARA features employed by the licensee are: dose rate surveys; an assessment of the N^{16} effects on plant and site dose rates; control of access to areas affected by N^{16} dose rate increases and limiting the number of people exposed; cessation of hydrogen addition when access to N^{16} affected areas is required; isotope scans on main steam lines to verify nuclides of interest; and comparison of background dose rates during normal operations and hydrogen injection mode dose rates. The licensee has additionally identified the need to evaluate extra shielding utilization and to modify operating procedures to further reduce doses. Site dose rate increases were determined to be on the order of 17% over single unit operation, which in turn is only a small fraction of dose rate increases observed with two unit operations. Specific plant areas were identified for additional limits on access. The licensee has identified alternative measures to identify fuel element failures during high power operations with full hydrogen additions.

In order for the NRC staff to evaluate the radiological impact of the long-term hydrogen injection program and to determine if additional or different radiological controls need to be considered, the licensee should perform a radiological assessment as recommended in Regulatory Guide 8.8, Sections C.1 and C.3 as follows:

Provide a summary radiological report to the NRC within 60 days of the completion of either a year's operation or completion of an operating cycle. (The report may be a planned contract report.) The report will contain the following:

(a) A summary/estimate of the additional manual dose incurred by major work function (e.g., as in NUREG-0713, Volume 3, Table 9) due to operation of the oxygen suppression system.
(b) An estimate of potential dose avoided for crack repairs and non-destructive analysis, both annually and over plant life.
(c) A summary of the value impact associated with operation of the system in terms of occupational dose (and other parameters as available, such as cost).
(d) A summary of the permanent radiation protection program changes needed to ensure that ALARA doses result from increased operational dose rates. This should include a discussion of such factors as access control procedures, posting changes, additional high radiation area controls, fuel element failure monitoring, H_2 system shutdown procedures for access, and other significantly affected radiation protection program parameters.
(e) Facility dose rate surveys which reflect typical operational conditions with and without hydrogen injection at high power. This should be a general area of survey of in-plant, on-site, and site boundary (e.g., fence) dose rates by such means as survey meters, μr meters, and area TLD's.

2.9 REMOVAL OF SNUBBER IN A SAFETY RELIEF VALVE DISCHARGE LINE

The licensee has requested in a letter dated March 24, 1983 that one snubber in the Target Rock Safety Relief Valve Discharge Line be removed. The bases for removal of the snubber are results of analyses performed to fulfill Mark I and I.E. Bulletin 79-14 requirements. The licensee stated that a more sophisticated analysis than previously performed negated the need for the subject snubber and that piping stresses on the SRV Discharge Line with the snubber removed are below the original design basis for Dresden Unit 2.

The staff concludes that the licensee has provided sufficient justification for removal of the subject snubber and approves the licensee's request to amend the Technical Specifications.

3.0 ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of an environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ACKNOWLEDGEMENTS

The following staff members have contributed to this evaluation:

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REFERENCES

1. Letter, T. J. Rausch to H. R. Denton, December 21, 1982 with the following attachments:

   (a) Dresden 2 and Cycle 9 Reload Discussion and Description of Technical Specification Changes.

   (b) Dresden 2, Cycle 9 Reload Analysis Report XN-NF-82-77(P), Rev. 1 dated November 1982.

   (c) Dresden 2, Cycle 9 Plant Transient Analysis Report, XN-NF-82-84(P), Rev. 1 dated November 1982.

   (d) Dresden Unit 2 LOCA Analysis Using the Exem Evaluation Model MAPLHGR Results, XN-NF-82-88, Rev. 1 dated November 1982.

   (e) Affidavit of R. B. Stout Attesting to the proprietary nature of XN-NF-81-75(P), dated December 1982.

   (f) Proposed Technical Specification Changes to DPR-19.

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