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One Full  
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Chicago, Illinois 60690

November 20, 1978

Director of Nuclear Reactor Regulation  
U.S. Nuclear Regulatory Commission  
Washington, DC 20555

Subject: Quad-Cities Station Unit 1  
Proposed Amendment to License and  
Appendix A, Technical Specifications,  
for Facility Operating License DPR-29  
to Support Reload No. 4  
NRC Docket No. 50-254

Dear Sir:

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Pursuant to 10 CFR 50.59, Commonwealth Edison proposes to amend the License and Appendix A, Technical Specifications, to Facility Operating License No. DPR-29 to support core reload No. 4 at Quad-Cities Station Unit 1. These changes are identified in Enclosure I and are based on plant analyses summarized in Enclosures II, III and IV.

It is planned, for Reload No. 4, to load 192 8x8R fuel assemblies, 4 of which will have developmental features designed for resistance to pellet-clad interaction (Barrier Lead Test Assemblies - BLTAs).

The primary reference for this Quad-Cities Unit 1 Reload 4 Cycle 5 licensing submittal is the General Electric Generic Reload Fuel Application (NEDE-24011). This new licensing submittal format contains similar technical information as previous submittals while deleting all explanatory text.

The attached enclosures that support this reload submittal are identified as follows: Enclosure II - "Supplemental Reload Licensing Submittal for Quad-Cities Nuclear Power Station Unit 1 Reload 4", NEDO-24145, 78NED283, September 1978; Enclosure III - "Quad-Cities Nuclear Power Station Unit 1 Reload 4 Supplemental Licensing Information for Barrier Lead Test Assemblies", NEDO-24147, 78NED285, September 1978; and Enclosure IV - "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146, 78NED284, September 1978.

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The significant changes in this transmittal include:

- a) a new MCPR Safety Limit of 1.07 as a result of the flatter local power (and CPR) distribution of the 8x8R design which, in turn, adversely affects the transition boiling probability distribution,
- b) a new LTPF of 3.00 for 8x8R fuel as a result of the increased active fuel length (145.24") and addition of a second water rod, and
- c) new MAPLHGR curves which reflect the improved flooding characteristics during a LOCA of the 8x8R design (which includes two alternate flow path holes drilled in the lower tie plate orifice nozzle.)

It should also be noted that a separate MCPR Limiting Condition of Operation has been specified for the Barrier Lead Test Assemblies (BLTAs). These four developmental bundles are virtually identical to the standard retrofit design (8DRB265-L) with the addition of:

- a) pellet-clad buffer materials (two bundles have copper barriers and two have zirconium liners),
- b) two segmented rods per BLTA (each consisting of four segments with hafnia-yttria end pellets), and
- c) pre-pressurization (3 atm. Helium).

Although these four bundles are neutronically treated the same as other reload fuel, a conservative treatment of the effects of pre-pressurization has been used for transient heat transfer assumptions and hence, the more restrictive MCPR LCO.

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A reanalysis of ECCS performance (Enclosure IV) for the limiting break size LOCA has resulted in relaxed MAPLHGR limits primarily due to the effects of drilled lower tie plates in the retrofit and BLTA reload fuel. The most notable feature of the new analysis is the utilization of Duane Arnold as the lead plant for several BWR 3's with loop selection logic. This approach allows units with similar ECCS characteristics to reference the full break spectrum analysis of the lead plant and necessitates detailed calculations for the limiting break size only (in this case the DBA value of 4.18 ft.<sup>2</sup>).

It should also be noted that Enclosure IV assumes only 156 drilled bundles which is conservative for the QCl C5 reload of 192 drilled bundles.

#### Conclusion

The safety impact of the new standard retrofit fuel design (which includes incorporation of two larger water rods, a refined enrichment distribution, and 6 inch regions of natural uranium at the bottom and top of the active fuel) has been generically evaluated and specifically evaluated for Quad-Cities 1 Reload 4 in Enclosure II. In all cases, the cumulative effect of the design changes has resulted in improved margin to established safety limits.

The impact of loading four special test assemblies has also been found to have little safety significance based on the evaluation presented in Enclosure III.

Although the increase enrichment in the central axial fuel zone is expected to increase operating MAPLHGR values, margin to limits is expected to improve since the LOCA analysis, incorporating the effects of drilled lower tie plates, increases the MAPLHGR limits proportionally greater than the expected increase in operating MAPLHGR values.

These proposed changes have received on-site and off-site review and approval.

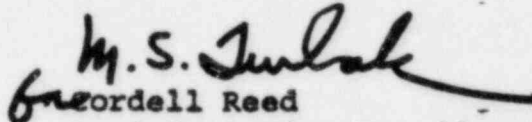
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Pursuant to 10 CFR 170, Commonwealth Edison has determined that the proposed amendment is Class III. As such, we have enclosed a fee remittance in the amount of \$4,000.00

For purposes of your schedule, the projected startup date for this unit is approximately 90 days from the date of this transmittal.

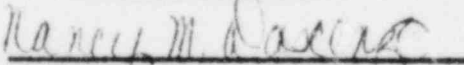
Three (3) signed originals and thirty-seven (37) copies of this transmittal are provided for your use.

Very truly yours,

  
Freordell Reed  
Assistant Vice-President

Enclosures

SUBSCRIBED and SWORN to  
before me this 20 day  
of November, 1978.

  
\_\_\_\_\_  
Notary Public

ENCLOSURE I

QUAD-CITIES UNIT 1

NRC DOCKET NO. 50-254

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

Commonwealth Edison is authorized to operate Quad-Cities Unit No. 1 at power levels not in excess of 2511 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 46, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Restrictions

Reactor power level shall be limited to maintain pressure margin to the safety valve set points during the worst case pressurization transient. The magnitude of the power limitation, if any, and the point in the cycle at which it shall be applied is specified in the Reload No. 4 licensing submittal for Quad Cities Unit No. 1 (NEDO-24145). Subsequent operation in the coastdown mode is permitted based on the Generic Reload Fuel Application (Pg. 5-9 of NEDE-24011-A) and its subsequent approval (D. G. Eisenhower to R. Gridley letter dated May 12, 1978).

D. Equalizer Valve Restriction

The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

4. This license is effective as of the date of issuance, and shall expire at midnight, February 15, 2007.

Closures: Appendices A and B--  
Technical Specifications

Date of Issuance: December 14, 1972

FOR THE ATOMIC ENERGY COMMISSION

*A. Giambusio*  
A. Giambusio, Deputy Director  
for Reactor Projects  
Directorate of Licensing

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**D. Reactor Water Level (Shutdown Condition)**

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core.

curve in Figure 2.1-2, at which point the actual peaking factor value shall be used.

$$\begin{aligned} \text{LTPF} &= 3.06 \text{ (7 x 7 fuel assemblies)} \\ &\quad 3.03 \text{ (8 x 8 fuel assemblies)} \\ &\quad 3.00 \text{ (8x8R fuel assemblies)} \end{aligned}$$

2. **APRM Flux Scram Trip Setting (Refueling or Startup and Hot Standby Mode)**

When the reactor mode switch is in the Refuel or Startup Hot Standby position, the APRM scram shall be set at less than or equal to 15% of rated neutron flux.

3. **IRM Flux Scram Trip Setting**

The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

4. When the reactor mode switch is in the startup or run position, the reactor shall not be operated in the natural circulation flow mode.

**B. APRM Rod Block Setting**

The APRM rod block setting shall be as shown in Figure 2.1-1 and shall be:

$$S \leq (.65W + 43) (\text{LTPF/TPF})$$

The definitions used above for the APRM scram trip apply.

- C. Reactor low water level scram setting shall be  $\geq$  143 inches above the top of the active fuel at normal operating conditions.
- D. Reactor low water level ECCS initiation shall be 83 inches (+4 inches/-0 inch) above the top of the active fuel at normal operating conditions.
- E. Turbine stop valve scram shall be  $\leq$  10% valve closure from full open.
- F. Turbine control valve fast closure scram shall initiate upon actuation of the fast closure solenoid valves which trip the turbine control valves.
- G. Main steamline isolation valve closure scram shall be  $\leq$  10% valve closure from full open.
- H. Main steamline low-pressure initiation of main steamline isolation valve closure shall be  $\geq$  850 psig.

## 1.1 SAFETY LIMIT BASES

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a safety limit such that the minimum critical power ratio (MCPR) is no less than 1.07. MCPR > 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

### A. Reactor Pressure > 800 psig and Core Flow > 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the cladding and therefore elevated cladding temperature and the possibility of cladding failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables (Figure 2.1-3).

The safety limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the normal operating condition, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit, 1.07, is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state, including uncertainty in the boiling transition correlation (see e.g., Reference 1). Because the boiling transition correlation is based on a large quantity of full-scale data, there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.07 would not produce boiling transition.

However, if boiling transition were to occur, cladding perforation would not be expected. Cladding temperatures would increase to approximately 1100° F, which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR), where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without cladding perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit (MCPR) operation is constrained to a maximum LHGR: 17.5 kw/ft for 7 x 7 fuel and 13.4 kw/ft for 8 x 8 fuel. This constraint is established by Specifications 2.1.A.1 and 3.5.J. Specification 2.1.A.1 established limiting total peaking factors (LTPF) which constrain LHGR's to the maximum values at 100% power and established procedures for adjusting APRM scram



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settings which maintain equivalent safety margins when the total peak factor (TPF) exceeds the LTPF. Specification 3.5J established the LHGR maximum which cannot be exceeded under steady power operation.

**B. Core Thermal Power Limit (Reactor Pressure < 800 psia)**

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of  $28 \times 10^3$  lb/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus the bundle flow with a 4.56-psi driving head will be greater than  $28 \times 10^3$  lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. At 25% of rated thermal power, the peak powered bundle would have to be operating at 3.86 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

**C. Power Transient**

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel, which is 8 to 9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail. In addition, control rod scrams are such that for normal operating transients, the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked each refueling outage, and at least every 32 weeks, 50% are checked to assure adequate insertion times. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification, a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the design limit of MCPR = 1.07 is not exceeded. Thus, use of a 1.5-second limit provides additional margin.

The computer provided has a sequence annunciation program which will indicate the sequence in which scrams occur, such as neutron flux, pressure, etc. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C.2 will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core-cooling capability could lead to elevated cladding temperatures and cladding perforation. The core will be cooled sufficiently to prevent cladding melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

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An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of TPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum total peaking factor is greater than the limiting total peaking factor.

2. APRM Flux Scram Trip Setting (Refuel or Startup/Hot Standby Mode)

For operation in the Startup mode while the reactor is at low pressure, the APRM scram setting of 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15% APRM scram remains active until the mode switch is placed in the Run position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM Flux Scram Trip Setting

The IRM system consists of eight chambers, four in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are broken down into 10 ranges, each being one-half a decade in size.

The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on Range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to 1% of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

**B. APRM Rod Block Trip Setting**

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate to protect against the condition of an MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst-case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds the limiting total peaking factor, thus preserving the APRM rod block safety margin.

**C. Reactor Low Water Level Scram**

The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

**D. Reactor Low Low Water Level ECCS Initiation Trip Point**

The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel cladding temperature to well below the cladding melting temperature to assure that core geometry remains intact and to limit any cladding metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each emergency core cooling system component was established based on the reactor low water level scram setpoint. To lower the setpoint of the low water level scram would increase the capacity requirement for each of the ECCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet will not be set lower because of ECCS capacity requirements.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could lead to a loss of effective core cooling. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

**E. Turbine Stop Valve Scram**

The turbine stop valve closure scram trip anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.07 even during the worst-case transient that assumes the turbine bypass is closed.

**F. Turbine Control Valve Fast Closure Scram**

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass, i.e., it prevents MCPR from becoming less than 1.07 for this transient. For the load rejection from 100% power, the LHGR increases to only 106.5% of its rated value, which results in only a small decrease in MCPR.

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1.2/2.1 REACTOR COOLANT SYSTEM

SAFETY LIMIT

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

LIMITING SAFETY SYSTEM SETTING

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

SPECIFICATIONS

A. The reactor coolant system pressure shall not exceed 1325 psig at any time when irradiated fuel is present in the reactor vessel.

A. Reactor coolant high-pressure scram shall be  $\leq 1060$  psig.

B. Primary system safety valve nominal settings shall be as follows:

- 1 valve at 1115 psig<sup>(1)</sup>
- 2 valves at 1240 psig
- 2 valves at 1250 psig
- 4 valves at 1260 psig

<sup>(1)</sup>Target Rock combination safety/relief valve

The allowable setpoint error for each valve shall be  $\pm 1\%$ .

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Venturi tubes are provided in the main steamline as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst-case accident, main steamline break outside the drywell, this trip setting of 120% of rated steam flow, in conjunction with the flow limiters and main steamline valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500° F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines (reference SAR Sections 14.2.3.9 and 14.2.3.10).

Temperature-monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200° F is low enough to detect leaks of the order of 5 to 10 gpm; thus it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high-steam flow instrumentation discussed above, and for small breaks with the resulting small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High-radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 7 times normal background and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident (reference SAR Section 12.2.1.7).

Pressure instrumentation is provided which trips when main steamline pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the Refuel and Startup/Hot Standby modes this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valve to open. With the trip set at 850 psig, inventory loss is limited so that fuel is not uncovered and peak cladding temperatures are much less than 1500° F; thus, there are no fission products available for release other than those in the reactor water (reference SAR Section 11.2.3).

The RCIC and the HPCI high flow and temperature instrumentation are provided to detect a break in their respective piping. Tripping of this instrumentation results in actuation of the RCIC or of HPCI isolation valves. Tripping logic for this function is the same as that for the main steamline isolation valves, thus all sensors are required to be operable or in a tripped condition to meet the single-failure criteria. The trip settings of 200° F and 300% of design flow and valve closure time are such that core uncover is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a one-out-of-two taken twice logic circuit. Unlike the reactor scram circuits, however, there is one trip system associated with each function rather than the two trip systems in the reactor protection system. The single-failure criteria are met by virtue of the fact that redundant core cooling functions are provided, e.g., sprays and automatic blowdown and high-pressure coolant injection. The specification requires that if a trip system becomes inoperable, the system which it activates is declared inoperable. For example, if the trip system for core spray A becomes inoperable, core spray A is declared inoperable and the out-of-service specifications of Specification 3.5 govern. This specification preserves the effectiveness of the system with respect to the single-failure criteria even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not approach 1.07. The trip logic for this function is one out of n; e.g., any trip on one of the six APRM's, eight IRM's, four SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure that the single-failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. This time period is only ~3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

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The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross core control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The APRM rod block function, which is set at 12% of rated power, is functional in the Refuel and Startup/Hot Standby modes. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby modes as the APRM flow-biased rod block does in the Run mode, i.e., it prevents MCPR from decreasing below 1.07 during control rod withdrawals and prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst-case single control rod withdrawal error has been analyzed, and the results show that with the specified trip settings, rod withdrawal is blocked before the MCPR reaches 1.07, thus allowing adequate margin (Reference 1).

Below 70% power, the worst-case withdrawal of a single control rod results in a MCPR greater than 1.07 without rod block action. Thus it is not required below this power level.

The IRM block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches 1.07.

A downscale indication on an APRM or IRM is an indication the instrument has failed or is not sensitive enough. In either case the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented. The downscale trips are set at 3/125 of full scale.

The SRM rod block with  $\leq 100$  CPS and the detector not fully inserted assures that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow determination of the cause of level increase and corrective action prior to automatic scram initiation.

For effective emergency core cooling for small pipe breaks, the HPCI system must function, since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met (reference SAR Section 6.2.6.3). The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and, when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a 15-minute delay before the air ejector off-gas isolation valve is closed. This delay is accounted for by the 30-minute holdup time of the off-gas before it is released to the chimney.

Both instruments are required for trip, but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the chimney release rate limit given in Specification 3.8.A.2 is not exceeded.

Four radiation monitors are provided in the reactor building ventilation ducts which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct. The trip logic is a one-out-of-two for each set, and each set can initiate a trip independent of the other set. Any upscale trip will cause the desired action. Trip settings of 2 mR/hr for monitors in the ventilation duct are based upon initiating normal ventilation isolation and standby gas treatment system operation so that the ventilation stack release rate limit given in Specification 3.8.A.3 is not exceeded. Two radiation monitors are provided on the refueling floor which initiate isolation of the reactor building and operation of the standby gas treatment systems. The trip logic is one-out-of-two. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation

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so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The instrumentation which is provided to monitor the postaccident condition is listed in Table 3.2-4. The instrumentation listed and the limiting conditions for operation on these systems ensure adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a definite knowledge of the conditions resulting from the accident; based on this information he can make logical decisions regarding postaccident recovery.

The specifications allow for postaccident instrumentation to be out of service for a period of 7 days. This period is based on the fact that several diverse instruments are available for guiding the operator should an accident occur, on the low probability of an instrument being out of service and an accident occurring in the 7-day period, and on engineering judgment.

The normal supply of air for the control room ventilation system comes from outside the service building. In the event of an accident, this source of air may be required to be shut down to prevent high doses of radiation in the control room. Rather than provide this isolation function on a radiation monitor installed in the intake air duct, signals which indicate an accident, i.e., high drywell pressure, low water level, main steamline high flow, or high radiation in the reactor building ventilation duct, will cause isolation of the intake air to the control room. The above trip signals result in immediate isolation of the control room ventilation system and thus minimize any radiation dose.

References

1. GE Topical Report NEDO-24145, "General Electric Boiling Water Reactor Reload No. 4 Licensing Submittal for Quad-Cities Nuclear Power Station (Unit 1)", Section 6.3.3.2, September, 1978.

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TABLE 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable or Tripped Instrument Channels per Trip System <sup>(1)</sup>	Instrument	Trip Level Setting
2	APRM upscale (flow bias) <sup>(7)</sup>	$\leq 0.650W + 43^{(2)}$
2	APRM upscale (Refuel and Startup/Hot Standby mode)	$\leq 12/125$ full scale
2	APRM downscale <sup>(7)</sup>	$\geq 3/125$ full scale
1	Rod block monitor upscale (flow bias) <sup>(7)</sup>	$\leq 0.650W + 42^{(2)}$
1	Rod block monitor downscale <sup>(7)</sup>	$\geq 3/125$ full scale
3	IRM downscale <sup>(3) (8)</sup>	$\geq 3/125$ full scale
3	IRM upscale <sup>(8)</sup>	$\leq 108/125$ full scale
2 <sup>(5)</sup>	SRM detector not in Startup position <sup>(4)</sup>	$\geq 2$ feet below core center-line
3	IRM detector not in Startup position <sup>(8)</sup>	$\geq 2$ feet below core center-line
2 <sup>(5) (6)</sup>	SRM upscale	$\leq 10^5$ counts/sec
2 <sup>(5)</sup>	SRM downscale <sup>(9)</sup>	$\geq 10^2$ counts/sec
1	High water level in scram discharge volume	$\leq 25$ gallons

Notes

- For the Startup/Hot Standby and Run positions of the reactor mode selector switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks. IRM upscale and IRM downscale need not be operable in the Run position. APRM downscale, APRM upscale (low biased), RBM upscale, and RBM downscale need not be operable in the Startup/Hot Standby mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to 7 days provided that during that time the operable system is functionally tested immediately and daily thereafter. If this condition lasts longer than 7 days the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.
- W is the reactor recirculation loop flow in percent. Trip level setting is in percent of rated power (2511 MWt).
- IRM downscale may be bypassed when it is on its lowest range.
- This function is bypassed when the count rate is  $\geq 100$  CPS.
- One of the four SRM inputs may be bypassed.
- This SRM function may be bypassed in the higher IRM ranges (ranges 8, 9, and 10) when the IRM upscale rod block is operable.
- Not required to be operable while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
- This IRM function occurs when the reactor mode switch is in the Refuel or Startup/Hot Standby position.
- This trip is bypassed when the SRM is fully inserted.



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- c. the operating power level shall be limited so that the MCPR will remain above 1.07 assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

<i>% Inserted From Fully Withdrawn</i>	<i>Average Scram Insertion Times (sec)</i>
5	0.375
20	0.900
50	2.00
90	3.50

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

<i>% Inserted From Fully Withdrawn</i>	<i>Average Scram Insertion Times (sec)</i>
5	0.398
20	0.954
50	2.12
90	3.80

2. The maximum scram insertion time for 90% insertion of any operable control rods shall not exceed 7 seconds.
3. If Specification 3.3.C.1 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.
4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be con-

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

2. Following a controlled shutdown of the reactor, but not more frequently than 16 weeks nor less frequently than 32-week intervals, 50% of the control rod drives in each quadrant of the reactor core shall be measured for the scram times specified in Specification 3.3.C. All control rod drives shall have experienced scram test measurements each year. Whenever all of the control rod drive scram times have been measured, an evaluation shall be made to

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- b. an end-of-cycle delayed neutron fraction of 0.005,
- c. a beginning-of-life Doppler reactivity feedback,
- d. the rod scram insertion rate shown in Specification 3.3.C,
- e. the maximum possible rod drop velocity of 3.11 fps,
- f. the design accident and scram reactivity shape function, and
- g. the moderator temperature at which criticality occurs.

In most cases the worth of insequence rods or rod segments will be substantially less than 0.013  $\Delta k$ . Further, the addition of 0.013  $\Delta k$  worth of reactivity, as a result of a rod drop and in conjunction with the actual values of the other important accident analysis parameters described above, would most likely result in a peak fuel enthalpy substantially less than 280 cal/g design limit. However, the 0.013  $\Delta k$  limit is applied in order to allow room for future reload changes and ease of verification without repetitive technical specification changes.

Should a control drop accident result in a peak fuel energy content of 280 cal/g, fewer than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in an offsite dose well below the guideline value of 10 CFR 100. For 8 x 8 fuel, fewer than 850 rods are conservatively estimated to perforate, with nearly the same consequences as for the 7 x 7 fuel case because of the rod power differences.

The rod worth minimizer provides automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences (reference SAR Section 7.9). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out of service when required, a licensed operator or other qualified technical employee can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

4. The source range monitor (SRM) system performs no automatic safety system function, i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron levels. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of  $10^{-8}$  of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's is provided as an added conservatism.
5. The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated **single control rod** could result in one of more fuel rods with MCPR's less than 1.07. During use of such patterns, it is judged that testing of the RBM system to assure its operability prior to withdrawal of such rods will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

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C. Scram Insertion Times

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than 1.0%. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than 1.0%. Reference 1 shows the control rod scram reactivity used in analyzing the transients. Reference 1 should not be confused with the total control rod worth, 18%  $\Delta k$ , as listed in some amendments to the SAR. The 18%  $\Delta k$  value represents the amount of reactivity available for withdrawal in the cold clean core, whereas the control rod worths shown in Reference 1 represent the amount of reactivity available for insertion (scram) in the hot operating core. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no more than 10% of the operable rods to have long scram times. In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid deenergizes. Approximately 200 milliseconds later, control rod motion begins. The time to deenergize the pilot valve scram solenoids is measured during the calibration tests required by Specification 4.1. The 200 milliseconds are included in the allowable scram insertion times specified in Specification 3.3.C.

The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be scram tested at increasing intervals following a shutdown.

Scram times of new drives are approximately 2.5 to 3 seconds; lower rates of change in scram times following initial plant operation at power are expected. The test schedule at increasing time intervals provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined below and judgment.

The probability that the mean 90% insertion time of a sample of 25 control rod drives will not exceed 0.25 seconds of the mean of all drives is 0.99 at a risk of 0.01. If the mean time exceeds this range or the mean 90% insertion time is greater than 3.5 seconds, an additional sample of drives will be measured to verify the mean performance.

Since the differences between the expected observed mean insertion time and the limit of Specification 3.3.C greatly exceed the expected range, this sampling technique gives assurance that the limits of Specification 3.3.C will not be exceeded. As further assurance that the limits of Specification 3.3.C will not be exceeded, all operable drives will be scram tested to determine compliance to Specification 3.3.C if the enlarged sample of 50 control rods exceeds 4.25 seconds. The 0.75 second margin to the limit is greater than the maximum expected deviation from the mean and therefore gives assurance that the mean will not exceed the limit of Specification 3.3.C. In addition, 50% of the control rods will be checked every 16 weeks to verify the performance and for correlation with the sampling program.

The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive exceeding the expected range of scram performance will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The numerical values assigned to the predicted scram performance are based on the analysis of the Dresden 2 startup data and of data from other BWR's such as Nine Mile Point and Oyster Creek.

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2 hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

$$LHGR_{max} = LHGR_d [1 - (\Delta P/P)_{max}(L/L_T)]$$

$LHGR_d$  = design LHGR

where:

= 17.5 kW/ft, 7 x 7 fuel assemblies

= 13.4 kW/ft, 8 x 8 fuel assemblies

$(\Delta P/P)_{max}$  = maximum power spiking penalty

= .035 initial core fuel

= .029 reload 1, 7 x 7 fuel

= .022 reload, 8 x 8 fuel

= .028 reload 1, mixed oxide fuel  
= .000 reload 8 x 8 fuel assemblies

$L_T$  = total core length

= 12 feet

$L$  = Axial distance from bottom of core

K. Minimum Critical Power Ratio (MCPR)

During steady-state operation MCPR shall be greater than or equal to

1.23 (7 x 7 fuel)

1.29 (8 x 8 fuel)

1.32 (8 x 8 BLTA)

at rated power and flow. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within 2 hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. For core flows other than rated, these nominal values of MCPR shall be increased by a factor of  $k_r$ , where  $k_r$  is as shown in Figure 3.5-2.

K. Minimum Critical Power Ratio (MCPR)

The MCPR shall be determined daily during steady-state power operation above 25% of rated thermal power.

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### 3.5 LIMITING CONDITION FOR OPERATION BASES

#### A. Core Spray and LPCI Mode of the RHR System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss-of-coolant analytical methods described in General Electric Topical Report NEDO-20566 and the specific analysis in NEDO-24146, "Loss-of-Coolant Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations, September 1978" core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel cladding temperature to less than 2200°F, to assure that core geometry remains intact, to limit cladding metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above. No single failure of ECCS equipment occurring during a loss-of-coolant accident under these limiting conditions of operation will result in inadequate cooling of the reactor core.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Quad-Cities 1 and 2, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray cooling of the reactor core before the internal pressure has fallen to 90 psig.

The LPCI mode of the RHR system is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel cladding temperature. The LPCI mode of the RHR system in combination with the core spray subsystem provides adequate cooling for break areas of approximately 0.2 ft<sup>2</sup> up to and including 4.18 ft<sup>2</sup>, the latter being the double-ended recirculation line break with the equalizer line between the recirculation loops, based without assistance from the high-pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference 1. Using the results developed in this reference, the repair period is found to be less than half the test interval. This assumes that the core spray subsystems and LPCI constitute a one-out-of-two system, however, the combined effect of the two systems to limit excessive cladding temperature must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 30 days, and this specification is within this period. For multiple failures, a shorter interval is specified, to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in Reference 1 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray subsystem and the entire LPCI mode of the RHR system are available should the need for core cooling arise. To assure that the remaining core spray, the LPCI mode of the RHR system, and the diesel generators are available, they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves and diesel generators. Based on judgments of the reliability of the remaining systems, i.e., the core spray and LPCI, a 7-day repair period was obtained.

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Should the loss of one RHR pump occur, a nearly full complement of core and containment cooling equipment is available. Three RHR pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, which will be demonstrated to be operable, a 30-day repair period is justified. If the LPCI mode of the RHR system is not available, at least two RHR pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

**B. RHR Service Water**

The containment cooling mode of the RHR system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and is therefore more than ample to provide the required heat-removal capability (reference SAR Section 5.2.3.2).

The containment cooling mode of the RHR system consists of two loops, each containing two RHR service water pumps, one heat exchanger, two RHR pumps, and the associated valves, piping, electrical equipment, and instrumentation. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability, as any one of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a 30-day repair period is adequate. Loss of one loop of the containment cooling mode of the RHR system leaves one remaining system to perform the containment cooling function. The operable system is demonstrated to be operable each day when the above condition occurs. Based on the fact that when one loop of the containment cooling mode of the RHR system becomes inoperable, only one system remains, which is tested daily, a 7-day repair period was specified.

**C. High-Pressure Coolant Injection**

The high-pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI mode of the RHR system or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of offsite electrical power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled, thus no cladding damage occurs (reference SAR Section 6.2.5.3). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

**D. Automatic Pressure Relief**

The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem. They enable the core spray subsystem or LPCI mode of the RHR system to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the core spray subsystems or LPCI mode of the RHR system. The core spray subsystem and the LPCI mode of the RHR system provide sufficient flow of coolant to limit fuel cladding temperatures less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

Loss of 1 of the relief valves affects the pressure relieving capability and, therefore, a 7 day repair period is specified.

Loss of more than one relief valve significantly reduces the pressure relief capability, thus a 24-hour repair period is specified based on the HPCI system availability during this period.

**E. RCIC**

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. Under these conditions the pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

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H. Condensate Pump Room Flood Protection

See Specification 3.5.H.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design-basis loss-of-coolant accident will not exceed the 2200°F limit specified in the 10 CFR 50 Appendix K considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat-generation rate of all the rods of a fuel assembly at any axial location and is only secondarily dependent on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak cladding temperature by less than ±20° F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the limit. The maximum average planar LHGR's shown in Figure 3.5-1 are based on calculations employing the models described in Reference 2.

J. Local LHGR

This specification assures that the maximum linear heat-generation rate in any rod is less than the design linear heat-generation rate even if fuel pellet densification is postulated. The power spike penalty specified is based on that presented in Reference 3 and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with a 95% confidence that no more than one fuel rod exceeds the design linear heat-generation rate due to power spiking. An irradiation growth factor of 0.25% was used as the basis for determining  $\Delta/P$  in accordance with References 4 and 5.

K. Minimum Critical Power Ratio (MCPR)

The steady state values for MCPR specified in this specification were selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state as well as uncertainties in the critical power correlation itself. These values also assure that operation will be such that the initial condition assumed for the LOCA analysis, an MCPR of 1.18, is satisfied. For any of the special set of transients or disturbances caused by single operator error or single equipment malfunction, it is required that design analyses initialized at this steady-state operating limit yield a MCPR of not less than that specified in Specification 1.1.A at any time during the transient, assuming instrument trip settings given in Specification 2.1. For analysis of the thermal consequences of these transients, the limiting value of MCPR stated in this specification is conservatively assumed to exist prior to the initiation of the transients. The results apply with increased conservatism while operating with MCPR's greater than specified.

The most limiting transients with respect to MCPR are generally:

- a) Rod withdrawal error
- b) Turbine trip without bypass
- c) Loss of feedwater heater

Several factors influence which of these transients results in the largest reduction in critical power ratio such as the specific fuel loading, exposure, and fuel type. The current cycles reload licensing submittal specifies the limiting transients for a given exposure increment for each fuel type. The values specified as the Limiting Condition of Operation are conservatively chosen as the most restrictive over the entire cycle for each fuel type.

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For core flow rates less than rated, the steady state MCPR is increased by the formula given in the specification. This assures that the MCPR will be maintained greater than that specified in Specification 1.1.A even in the event that the motor-generator set speed controller causes the scoop tube positioner for the fluid coupler to move to the maximum speed position.

References

1. I. M. Jacobs and P. W. Marritt, GE Topical Report APED-5736, 'Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards,' April 1969.
2. "Loss-of-Coolant-Accident Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146, September 1978.
3. GE Topical Report NEDM-10735, 'Fuel Densification Effects on General Electric Boiling Water Reactor Fuel,' Section 3.2.1, Supplement 6, August 1973.
4. J. A. Hinds, GE, Letter to V. A. Moore, USAEC, 'Plant Evaluation with GE GEGAP-III,' December 12, 1973.
5. USAEC Report, 'Supplement I to the Technical Report on Densification of General Electric Reactor Fuels,' December 14, 1973.



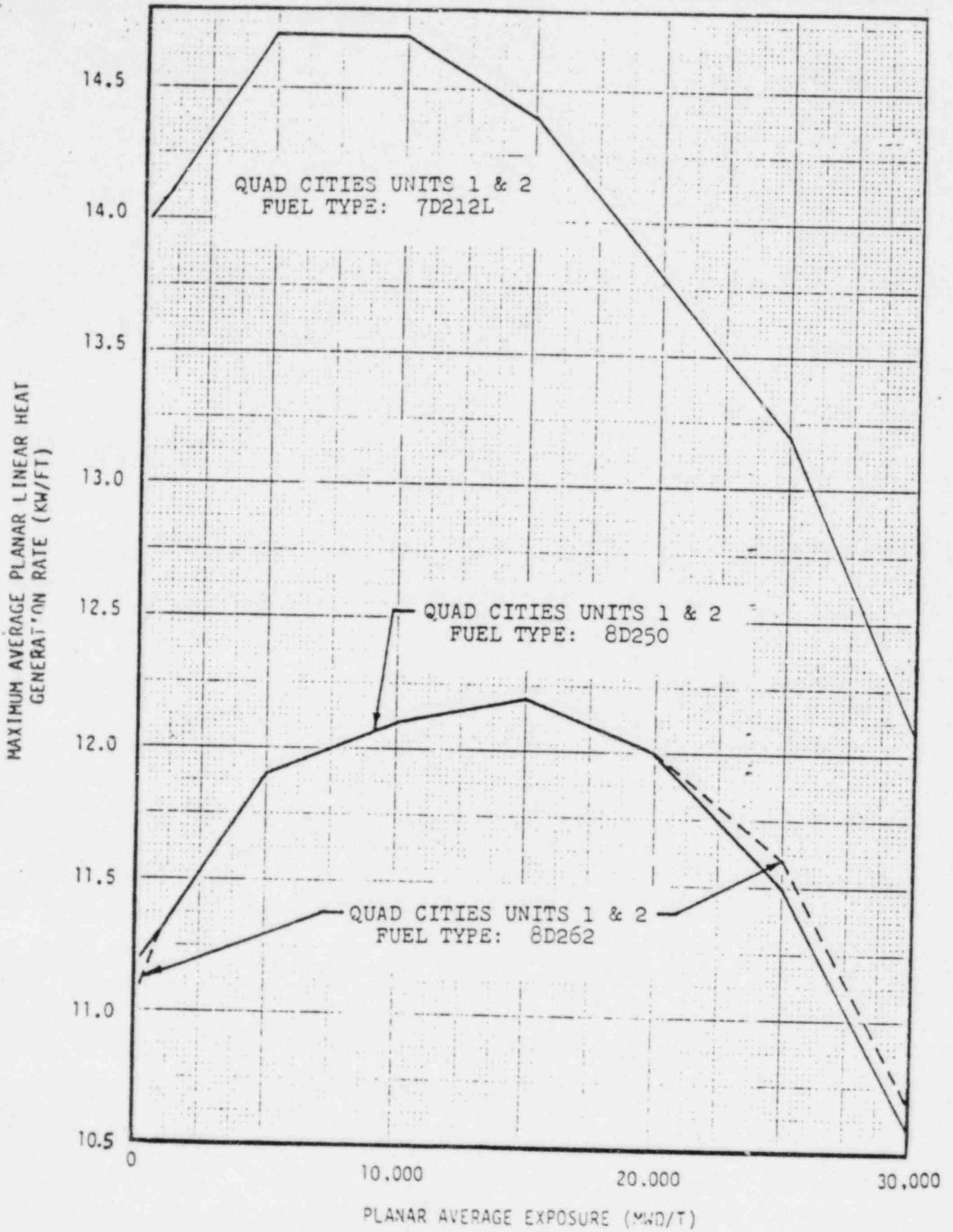


FIGURE 3.5-1

MAXIMUM AVERAGE PLANAR LINEAR  
HEAT GENERATION RATE (MAPLHGR)  
VS. PLANAR AVERAGE EXPOSURE

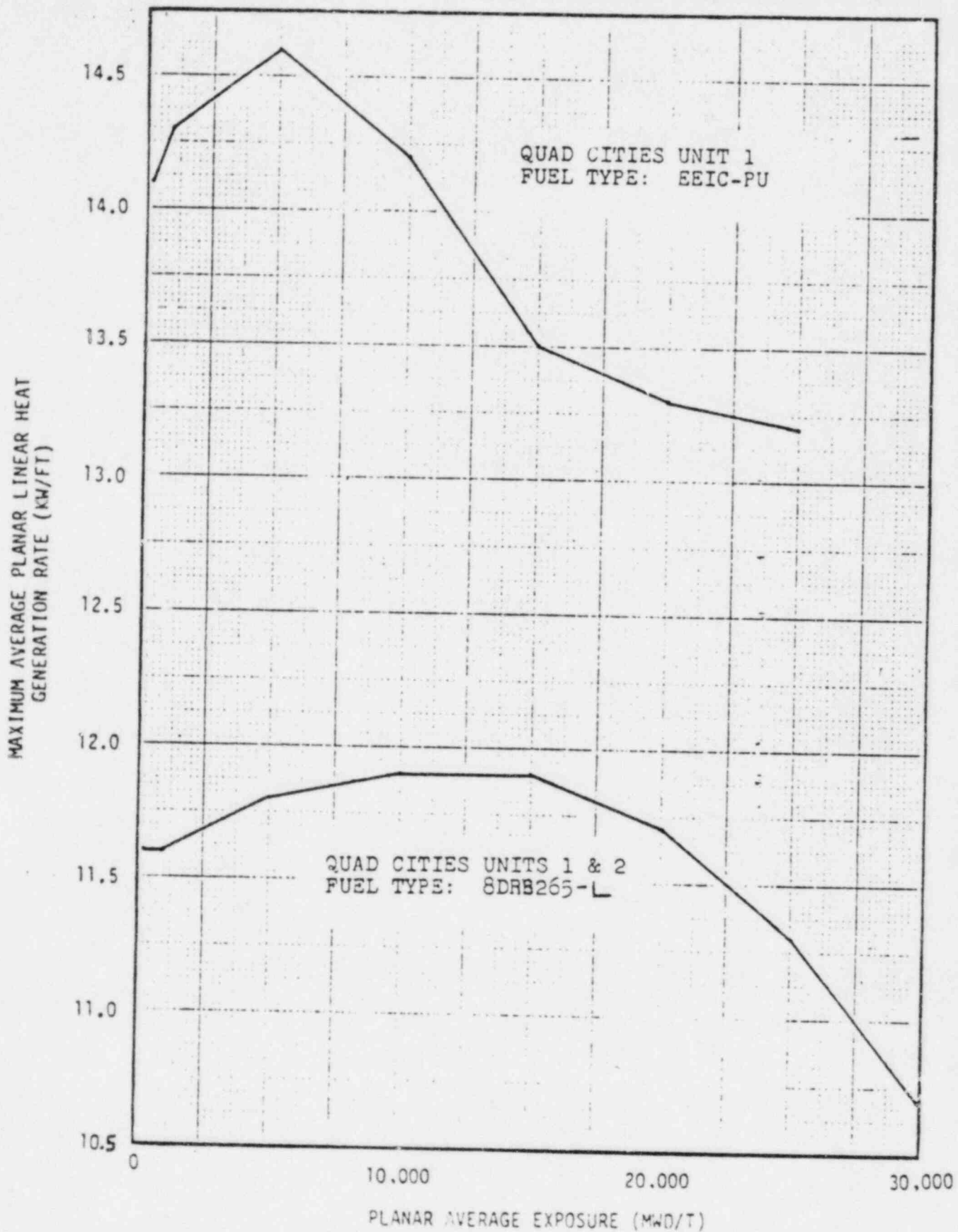


FIGURE 3.5-1

MAXIMUM AVERAGE PLANAR LINEAR  
HEAT GENERATION RATE (MAPLHGR)  
VS. PLANAR AVERAGE EXPOSURE

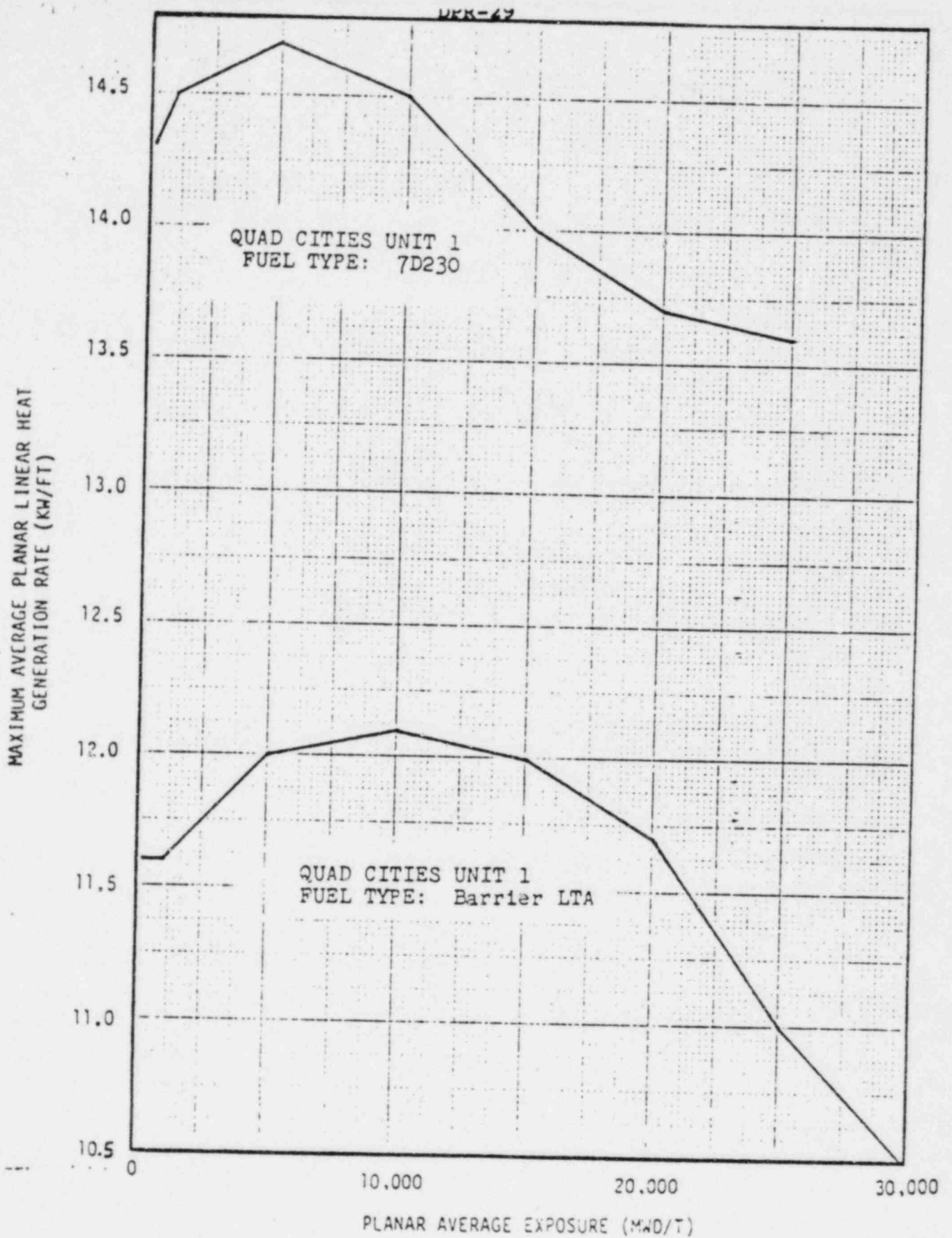


FIGURE 3.5-1

MAXIMUM AVERAGE PLANAR LINEAR  
HEAT GENERATION RATE (MAPLHGR)  
VS. PLANAR AVERAGE EXPOSURE

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- Both the sump and air sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 7 days.
- If the conditions in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

E. Safety and Relief Valves

- Prior to reactor startup for power operation, 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.
- If Specification 3.6.E.1 is not met, the reactor shall remain shut down until the condition is corrected or, if in operation, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be below 90 psig and 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", 1974 Edition, Summer 1975 Addenda (ASME Code Section XI).

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	Setpoint (psig)
1	1115 (1)
2	1240
2	1250
4	1260

The allowable setpoint error for each valve is  $\pm 1\%$ .

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

Number of Valves	Setpoint (psig)
1	$\leq 1115$ (1)
2	$\leq 1130$
2	$\leq 1135$

(1) Target Rock combination safety/relief valve.

F. Structural Integrity

The nondestructive inspections listed in Table 4.6-1 shall be performed as specified in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, Summer 1971 Addenda. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions will be reviewed with the NRC.

ENCLOSURE II

QUAD-CITIES UNIT 1

NRC DOCKET NO. 50-254

NEDO-24145  
78NED283  
Class I  
September 1978

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL  
FOR  
QUAD CITIES NUCLEAR POWER STATION  
UNIT 1 RELOAD 4

Prepared: *R. T. Hill*  
R.T. Hill, Senior Licensing Engineer  
Operating Licenses I

Approved: *J. E. Engel*  
J.E. Engel, Manager  
Operating Licenses I

NUCLEAR ENERGY PROJECTS DIVISION • GENERAL ELECTRIC COMPANY  
SAN JOSE, CALIFORNIA 95125

GENERAL  ELECTRIC

IMPORTANT NOTICE REGARDING  
CONTENTS OF THIS REPORT  
PLEASE READ CAREFULLY

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1. PLANT UNIQUE ITEMS (1.C)\*

- a. Plant parameter changes                      See Appendix A
- b. Loading Error                                      See Appendix A
- c. Loss-of-Coolant Accident Analysis            See Reference 1 (pg 5)
- d. Barrier Lead Test Assembly (BLTA)           See Reference 2 (pg 5)
- e. R (item 4)                                         Value shown includes effect of  $B_4C$  settling (0.0004  $\Delta k$ )

2. RELOAD FUEL BUNDLES (1.0, 3.3.1 and 4.0)

	<u>Fuel Type</u>	<u>Number</u>	<u>Number Drilled</u>
Irradiated	Initial (7DB212)	128	-
	Reload-1 (7DB230)	22	-
	(7DB230-STK)	1	-
	(7DB230-Pu)	5	-
	(8DB250)	36	-
	Reload-2 (8DB250)	104	-
	(8DB262)	52	-
New	Reload-3 (8DB250)	184	-
	Reload-4 (8DRB265L)	192	192
Total		724	192

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle exposure: 15,695 MWd/t.  
 Assumed reload cycle exposure: 16,100 MWd/t.  
 Core loading pattern: Figure 1.

4. CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C (3.3.2.1.1 AND 3.3.2.1.2)

BOC $k_{eff}$	
Uncontrolled	1.106
Fully Controlled	0.945
Strongest Control Rod Out	0.980
R, Maximum Increase in Cold Core Reactivity with Exposure Into Cycle, $\Delta k$	0.0024

\* ( ) refers to areas of discussion in "Generic Reload Fuel Application," NEDE-24011-P-A, August 1978.



5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

<u>PPM</u>	<u>Shutdown Margin (<math>\Delta k</math>)</u> <u>(20°C, Xenon Free)</u>
600	0.045

6. RELOAD UNIQUE TRANSIENT ANALYSIS INPUTS (3.3.2.1.5 AND 5.2)

	<u>EOC</u>	<u>EOC-1000 MWd/t</u>
Void Coefficient N/A*(c/% Rg)	-5.82/-7.28	-6.57/-8.21
Void Fraction (%)	34.49	34.49
Doppler Coefficient N/A (c/%°F)	-0.229/-0.217	-0.223/-0.212
Average Fuel Temperature (°F)	1203	1203
Scram Worth N/A (\$)	-41.29/-33.03	-39.22/-31.38
Scram Reactivity vs Time	Figure 2a	Figure 2b

7. RELOAD UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (5.2)

	<u>EOC 7x7</u>	<u>8x8/8xSR</u>
Peaking factors (local, radial and axial)	1.30	1.22
	1.56	1.84
	1.40	1.40
R-Factor	1.100	1.051
Bundle Power (MWt)	5.304	6.245
Bundle Flow (10 <sup>3</sup> lb/hr)	115.88	107.50
Initial MCPR	1.23	1.29

\*N = Nuclear Input Data  
A = Used in Transient Analysis

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure	Power (%)	Flow (%)	t (s)	Q/A (s)	P <sub>sl</sub> (psig)	P <sub>v</sub> (psig)	ΔCPR		Plant Response
								7x7	8x8/8x8R	
Load Rejection Without Bypass	EOC	99	100	264.4	110.7	1213	1248	0.16	0.22	Figure 3a
Turbine Trip Without Bypass	EOC - 1000 MWd/t	100	100	225.0	108.8	1214	1248			Figure 3b
Loss of 145°F Feedwater Heating	--	100	100	120.7	119.4	991	1045	0.15	0.18	Figure 4
Feedwater Controller Failure	EOC	100	100	155.3	107.5	1123	1158	0.08	0.11	Figure 5a
Feedwater Controller Failure	EOC - 1000 MWd/t	100	100	131.2	105.7	1117	1153			Figure 5b

10. LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY (5.2.1)

Rod Block Reading	Rod Position (Feet Withdrawn)	ΔCPR		MLHGR (Kw/ft)		Limiting Rod Pattern
		7x7	8x8/8x8R	7x7	8x8/8x8R	
104	3.5	0.10	0.16	16.34	15.30	Figure 6
105	3.5	0.10	0.16	16.34	15.30	Figure 6
106	4.0	0.11	0.18	16.72	15.86	Figure 6
*107	4.5	0.13	0.20	16.90	16.20	Figure 6
108	6.5	0.17	0.26	15.66	15.46	Figure 6
109	7.5	0.24	0.28	15.26	15.70	Figure 6

\*Indicates setpoint selected

11. OPERATING MCPR LIMIT (5.2)

1.29 (8x8/8x8R fuel)

1.23 (7x7 fuel)

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

<u>Transient</u>	<u>Power (%)</u>	<u>Core Flow (%)</u>	<u>P<sub>sl</sub> (psig)</u>	<u>P<sub>v</sub> (psig)</u>	<u>Plant Response</u>
MSIV Closure (Flux Scram)	100	100	1276	1310	Figure 7

13. STABILITY ANALYSIS RESULTS (5.4)

Decay Ratio: Figure 8

Reactor Core Stability:

Decay Ratio,  $x_2/x_0$  0.56

(100% Rod Line - Natural  
Circulation Power)

Channel Hydrodynamic Performance

Decay Ratio,  $x_2/x_0$   
(100% Rod Line - Natural  
Circulation Power)

8x8/8x8R channel 0.14

7x7 channel 0.04

14. LOSS-OF-COOLANT ACCIDENT RESULTS (5.5.2)

See Reference 1.

15. LOADING ERROR RESULTS (5.5.4)

Limiting Event: Misplaced bundle

MCPR: 1.07

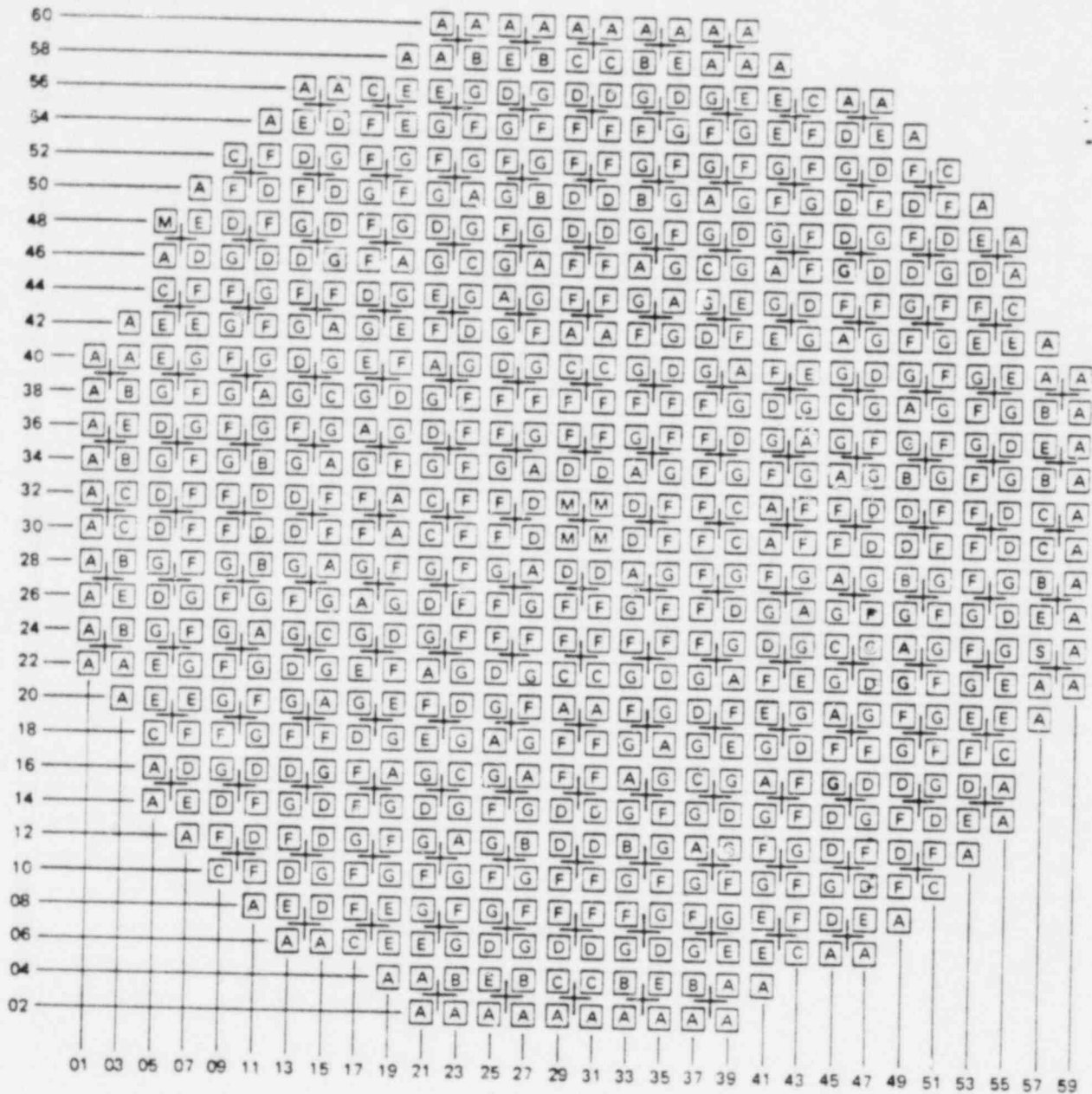
16. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

Maximum incremental control rod worth: ~~0.8%~~  $\Delta k$   
0.658%

AS  
11/14/78

REFERENCES:

- (1) "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad Cities Units 1,2 Nuclear Power Stations," NEDO-24146, September 1978.
- (2) "Quad Cities Nuclear Power Station Unit 1 Reload 4 Supplemental Licensing Information For Barrier Lead Test Assemblies," NEDO-24147, September 1978



FUEL TYPE	
A = INITIAL FUEL	D = RELOAD 2 (8DB250)
B = RELOAD 1 (TDB230) (M = MIXED OXIDE)	E = RELOAD 2 (8DB262)
C = RELOAD 1 (8DB250)	F = RELOAD 3 (8DB250)
	G = RELOAD 4 (8DRB265L)

Figure 1. Quad Cities Unit 1 Reload 4 Design Reference Core Loading

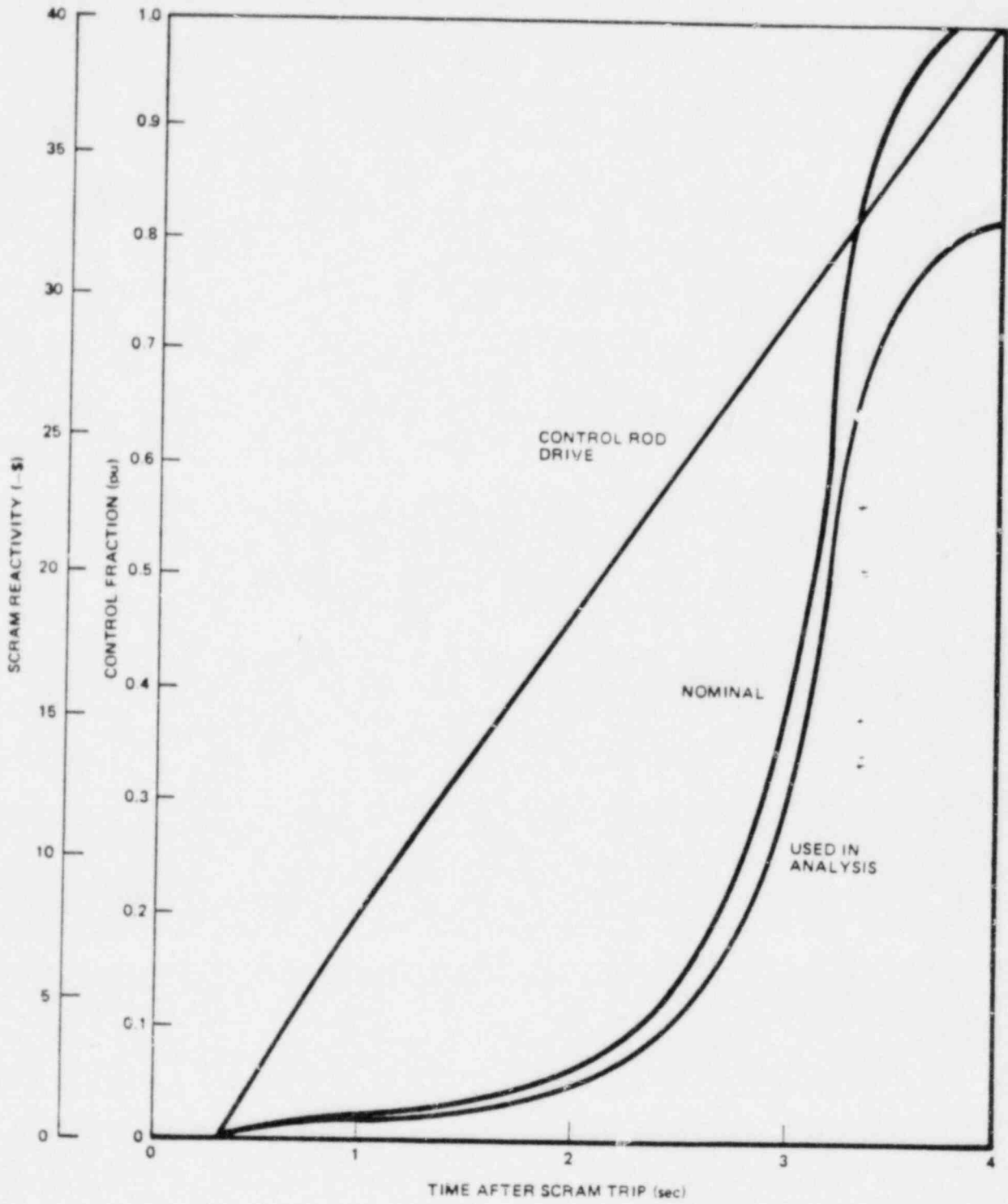


Figure 2a. Scram Reactivity and Control Rod Drive Specification vs Time (EOC5)

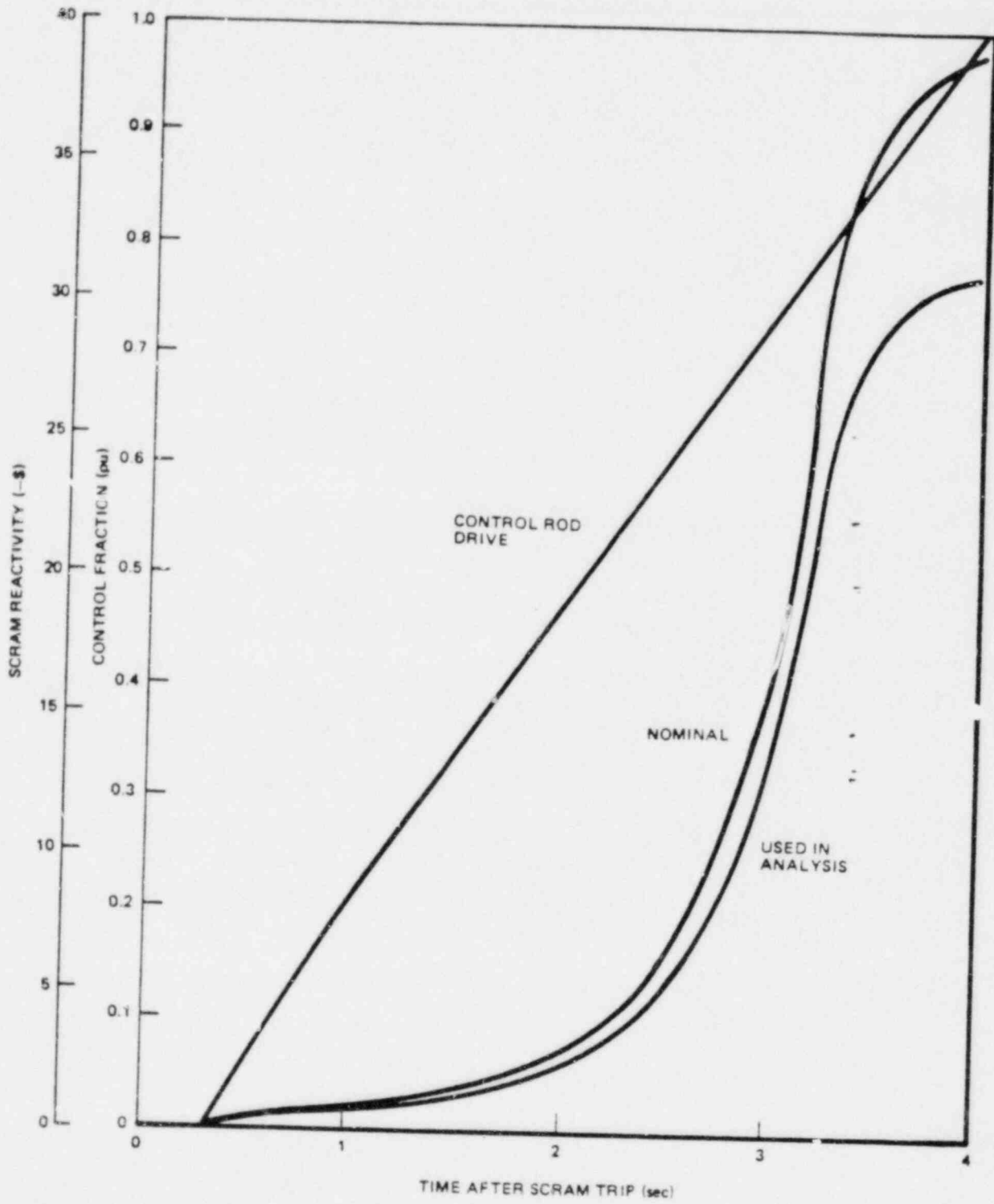


Figure 2b. Scram Reactivity and Control Rod Drive Specification vs Time (1000 MWd/t Before EOC5)

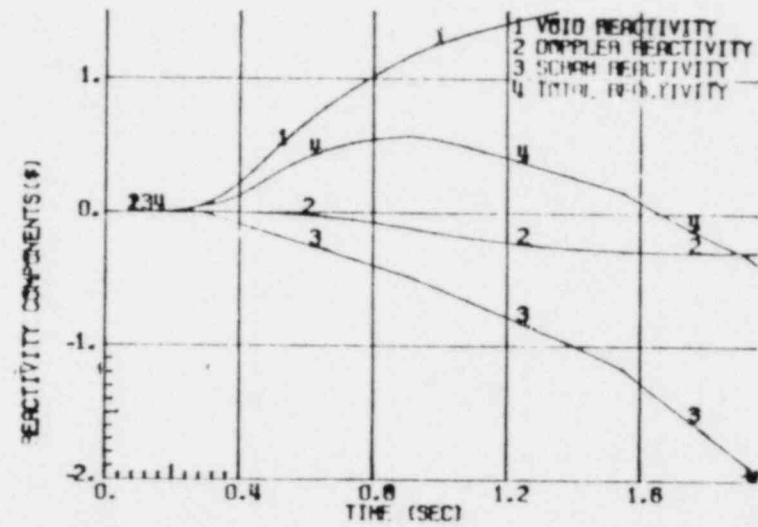
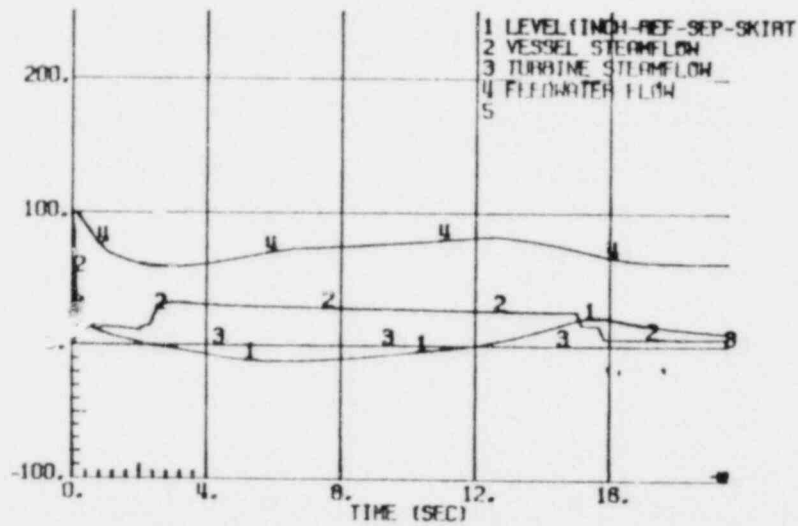
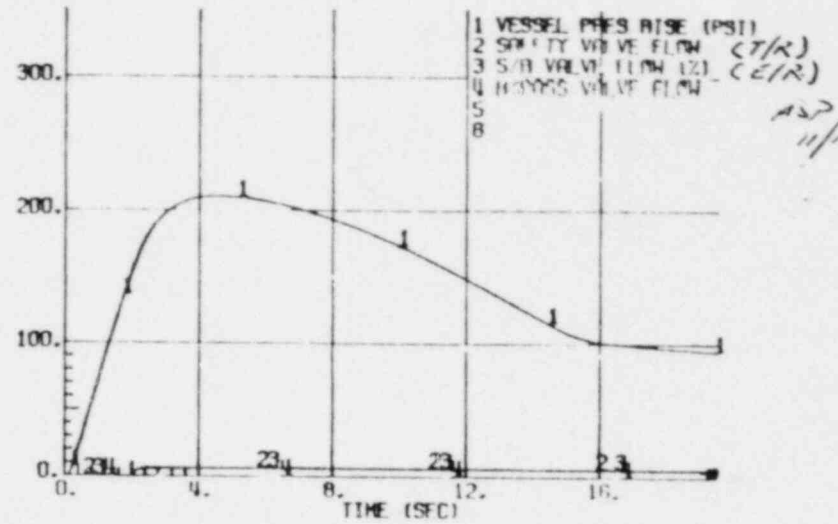
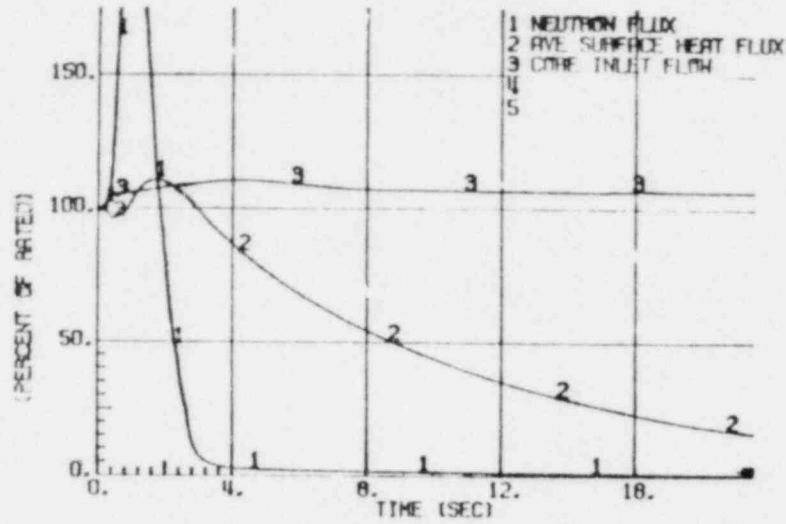


Figure 3a. Load Rejection Without Bypass, EOC, 99% Power



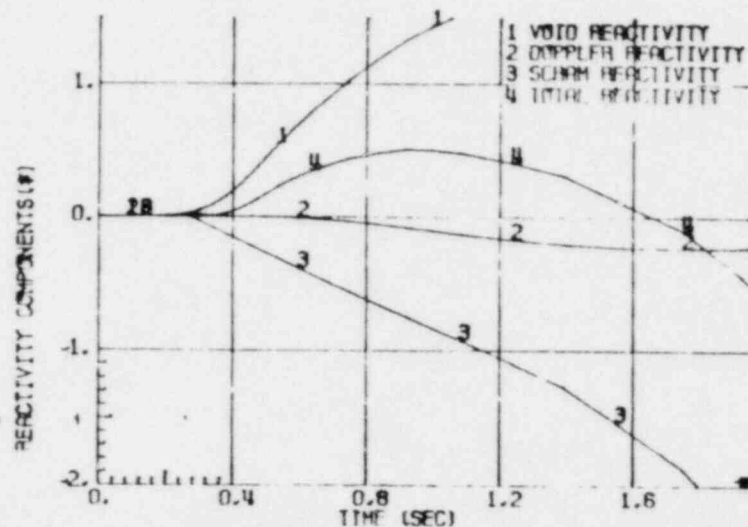
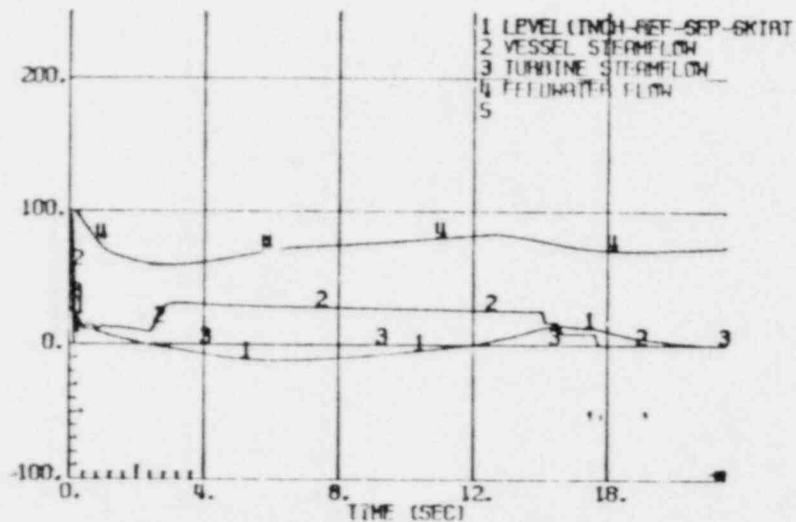
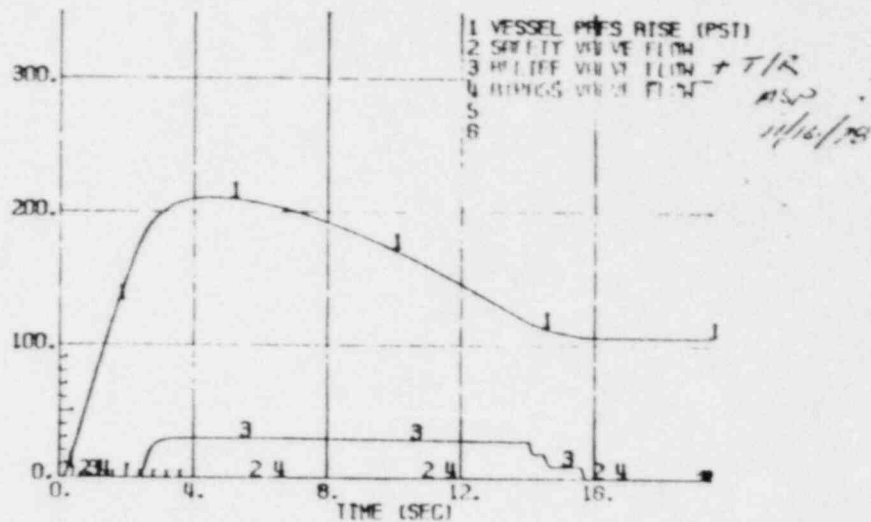
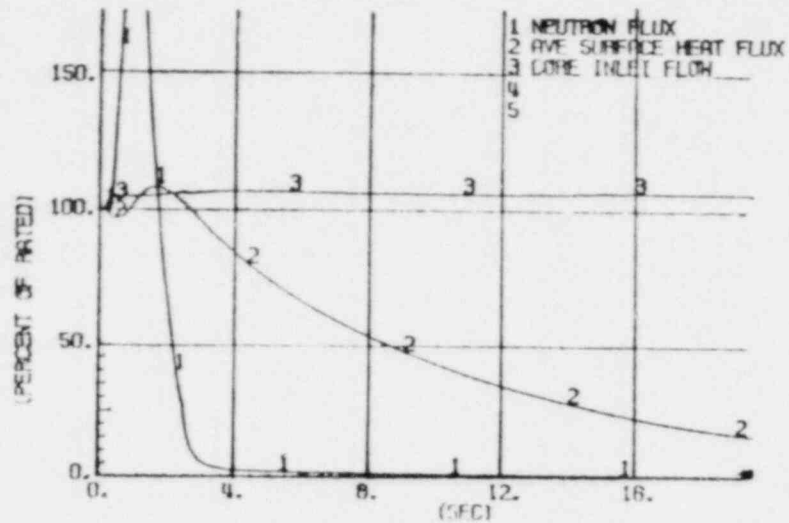


Figure 3b. Turbine Trip Without Bypass, EOC-1000 MWd/t, 100% Power

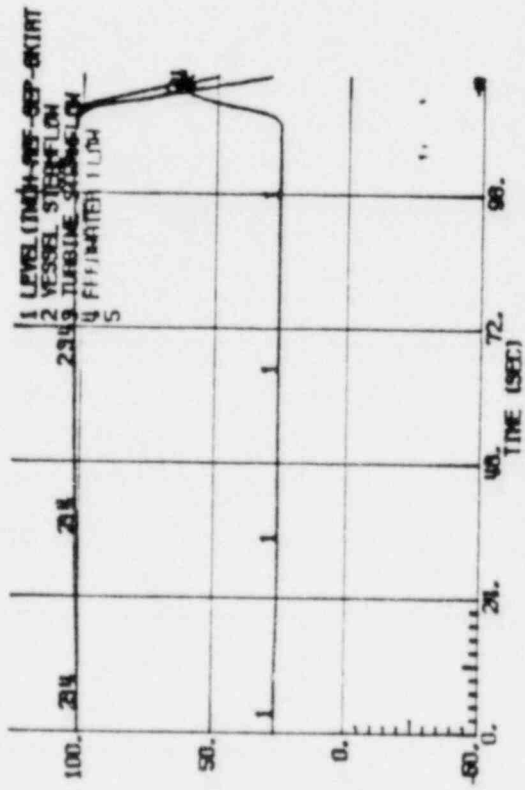
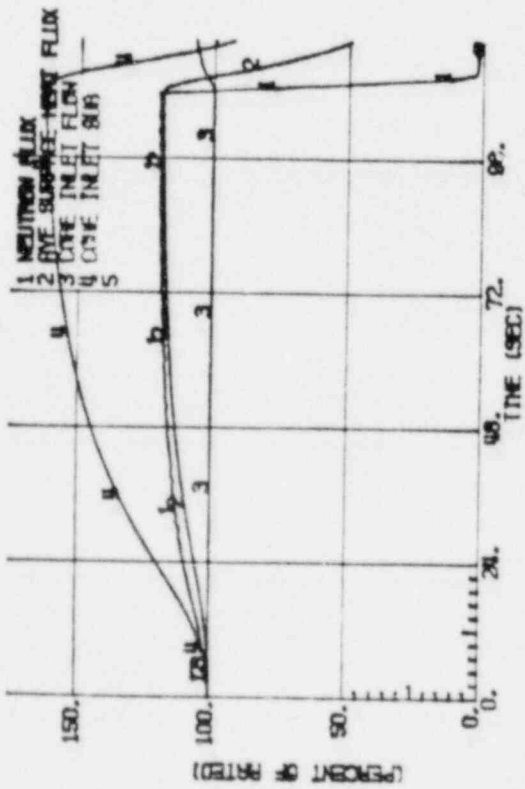
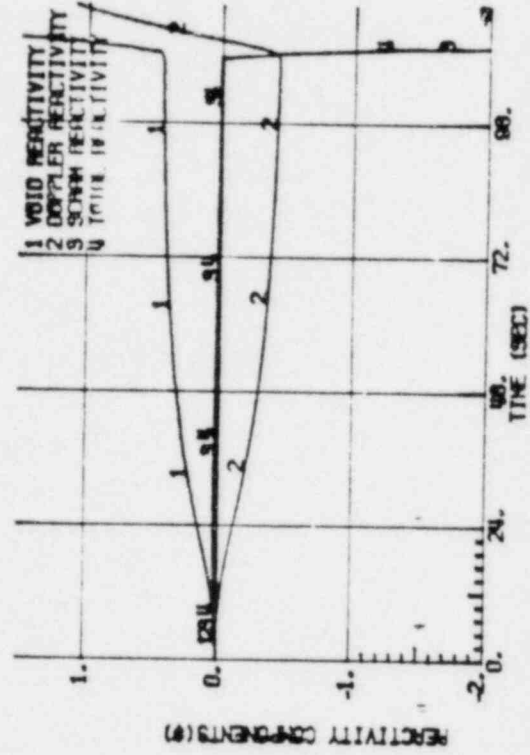
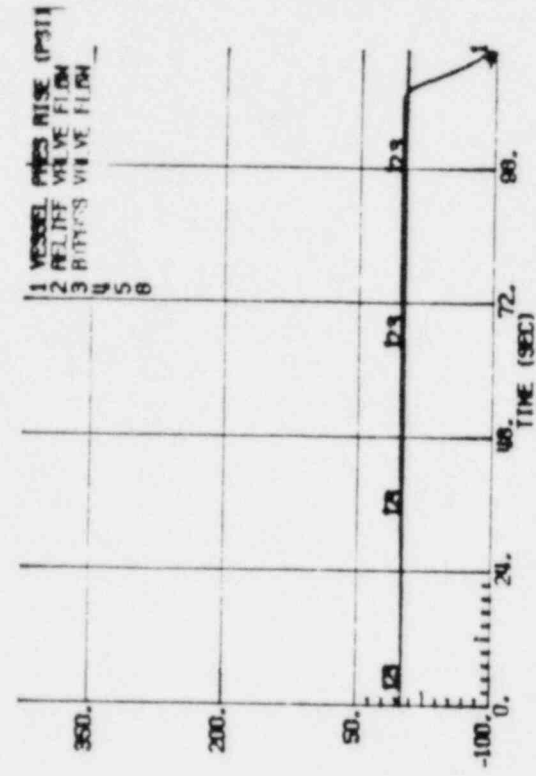


Figure 4. Loss of 145°F Feedwater Heating

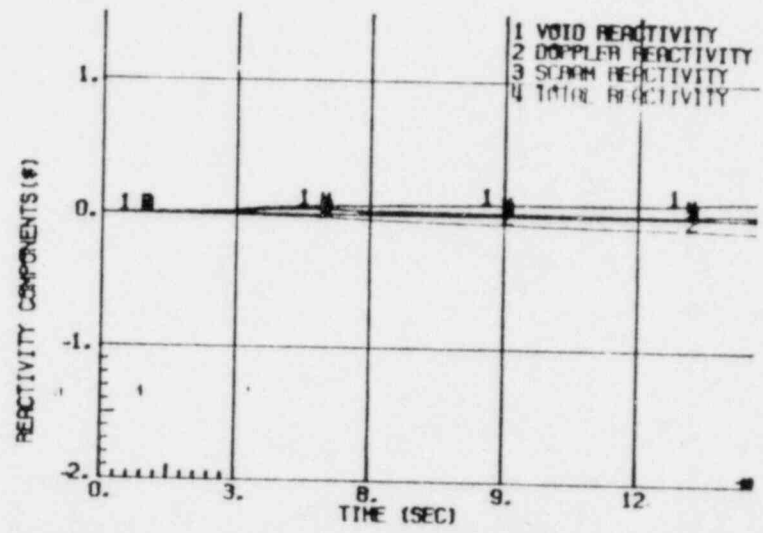
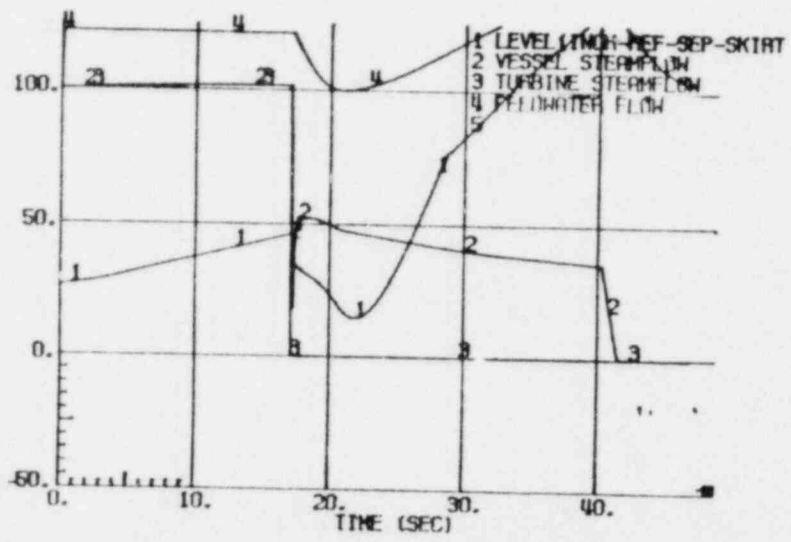
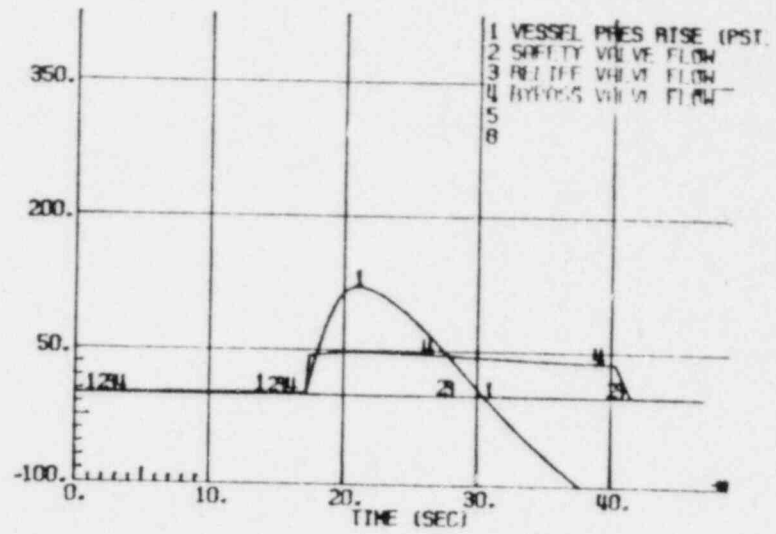
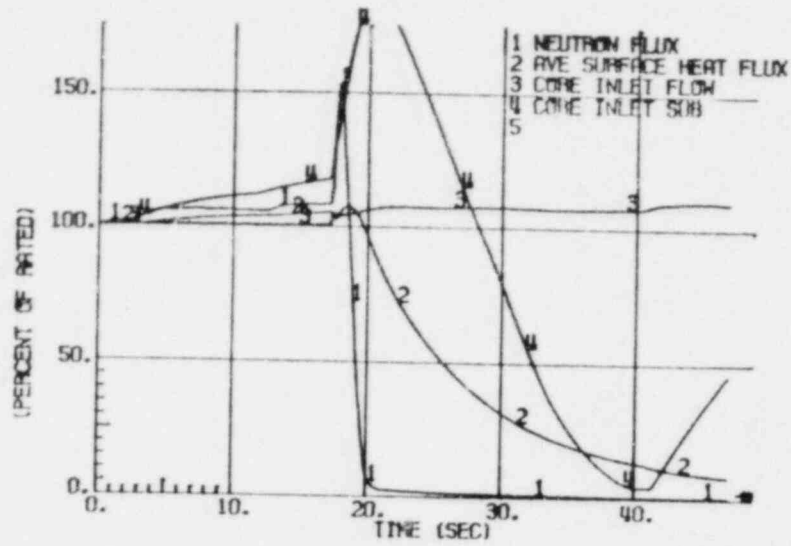


Figure 5a. Feedwater Controller Failure-Maximum Demand, EOC

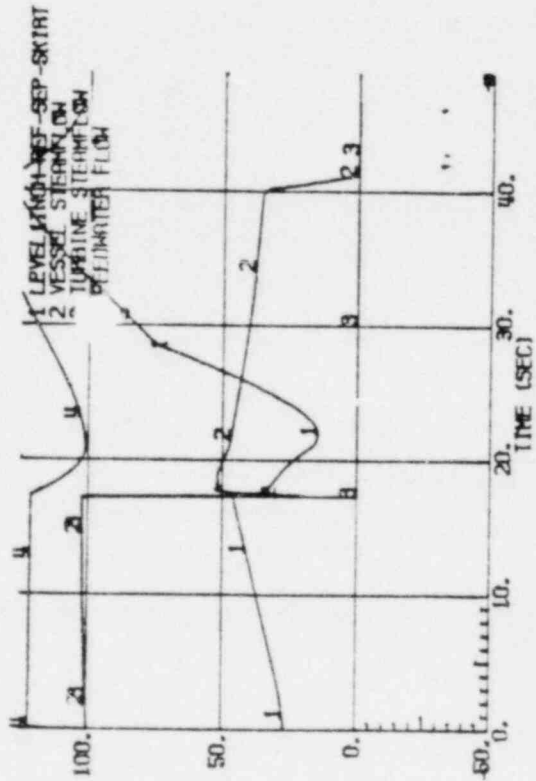
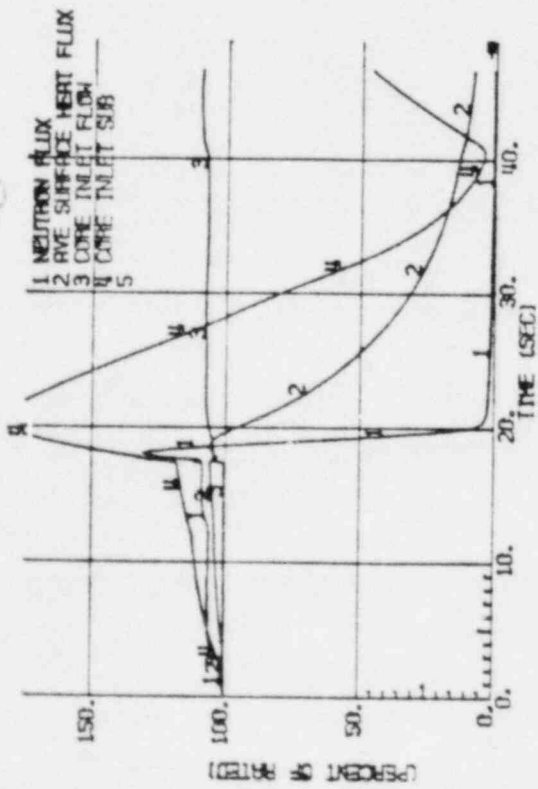
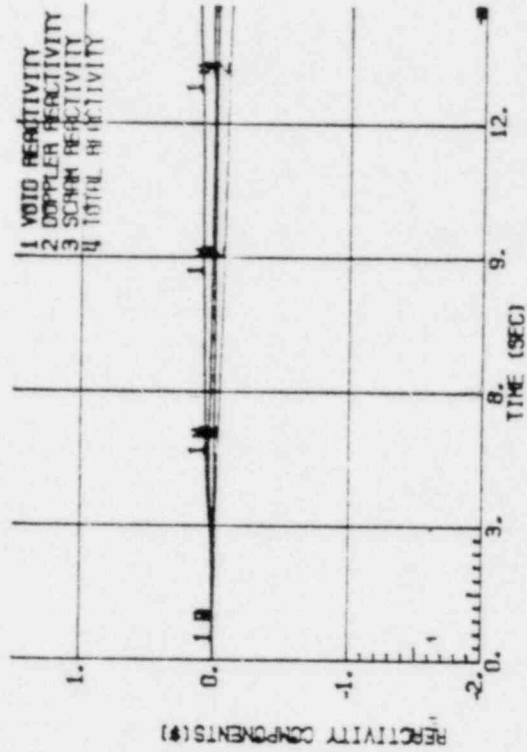
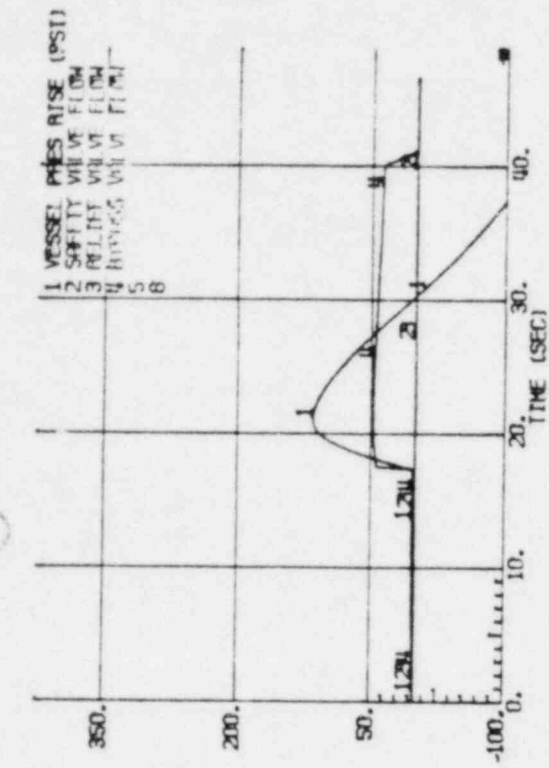


Figure 5b. Feedwater Controller Failure - Maximum Demand, EOC-1000 MWd/t

	02	06	10	14	18	22	26	30
59							8	
55				14		6		14
51			10		20		12	
47		14		6		10		6
41			20		34		36	
39		6		10		0		16
35	10		12		36		36	
31		18		6		16		6

- NOTES: 1. ROD PATTERN IS 1/4 CORE MIRROR SYMMETRIC UPPER LEFT QUADRANT SHOWN ON MAP  
 2. NUMBERS INDICATE NUMBER OF NOTCHES WITHDRAWN OUT OF 48. BLANK IS A WITHDRAWN ROD  
 3. ERROR ROD IS (11,11)

Figure 6. Limiting Rod Pattern For Rod Withdrawal Error

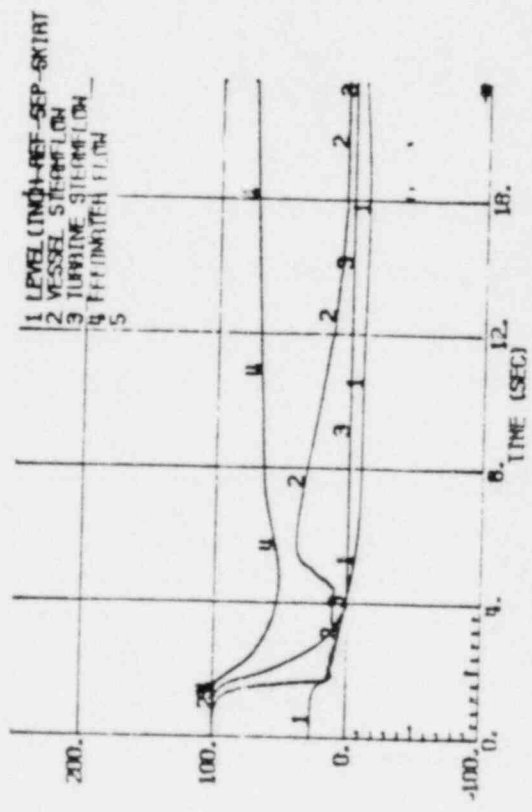
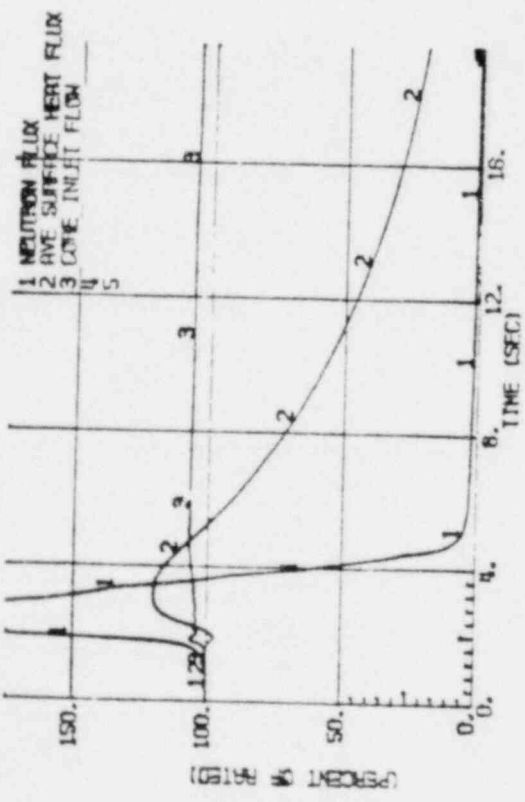
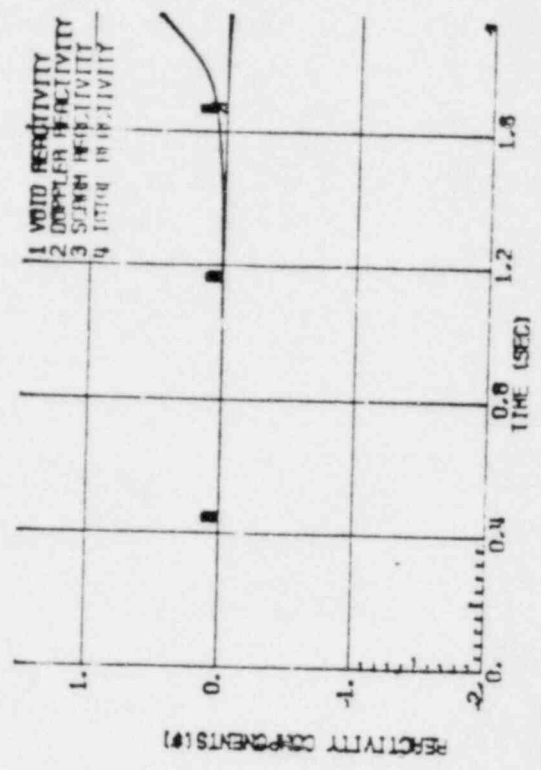
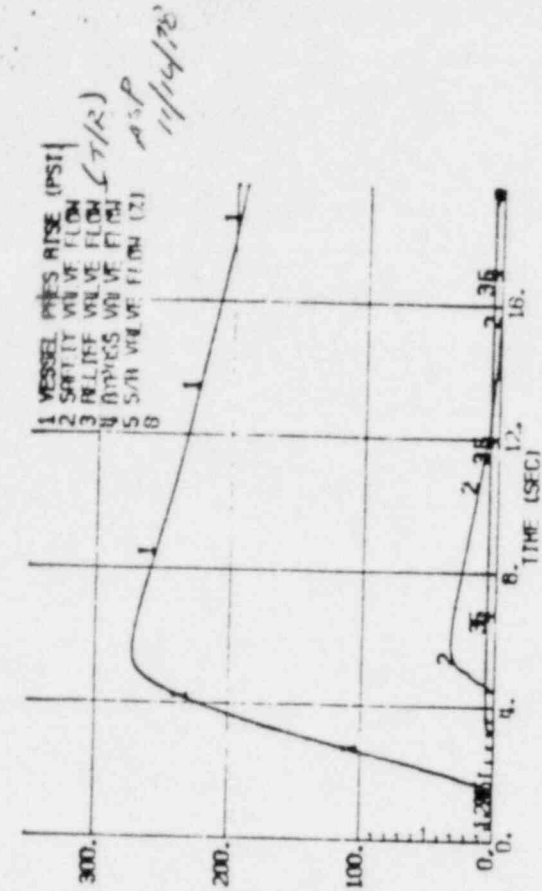


Figure 7. Plant Response to MSIV Closure

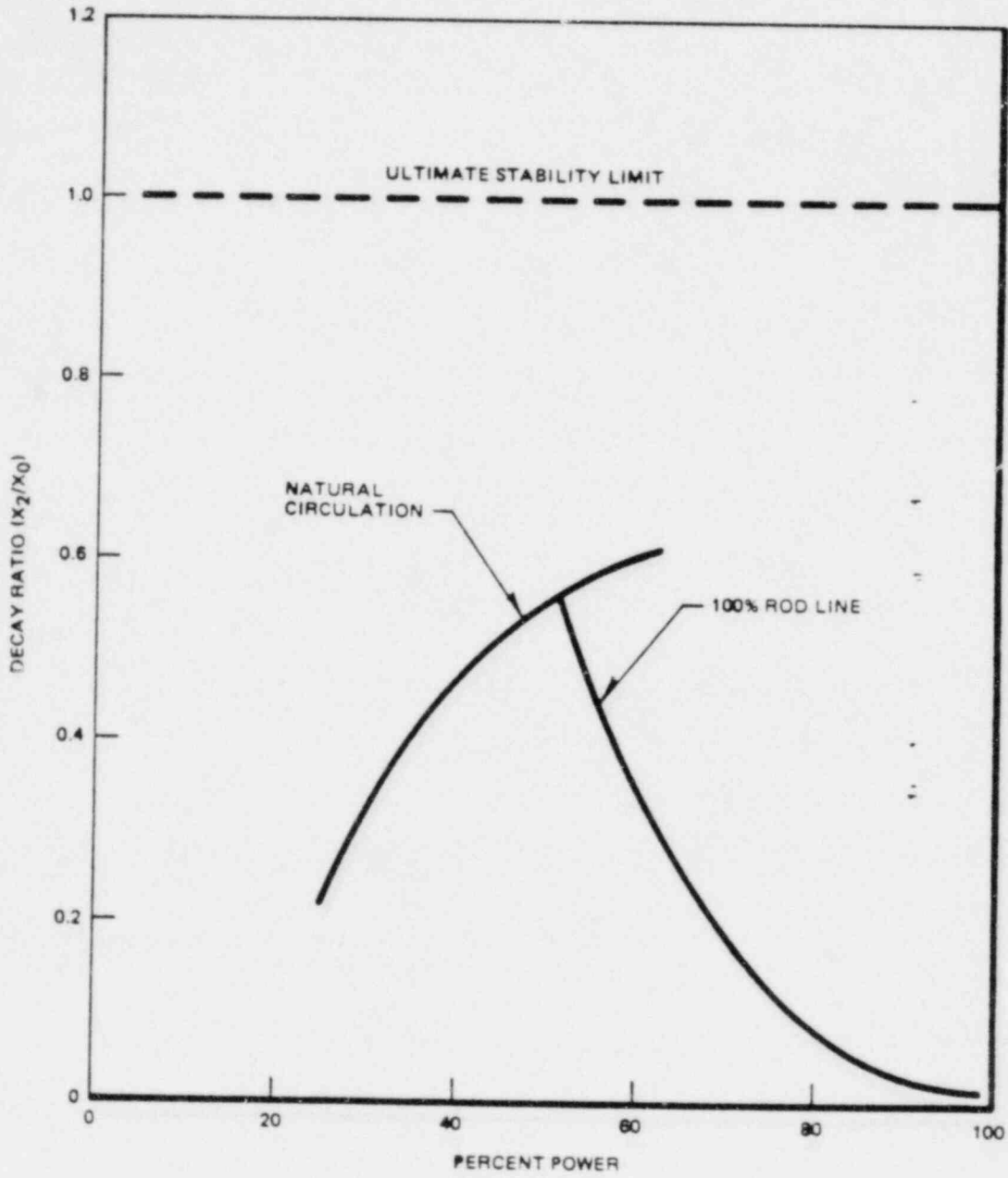


Figure 8. Decay Ratio

## APPENDIX A

## PLANT PARAMETER CHANGES

## Pressure Relief Systems (Table 5-4, pg 5-62, NEDO-24011)

Safety/Relief valve setpoint (psig)	11.15 + 1%
<i>4 RELIEF + 1</i> Safety/Relief Valve capacity (% rated steam flow)	27.8
<i>ASP</i> <i>11/16/78</i> Safety Valve capacity (% rated steam flow)	50

## Transient Operating Parameters (Table 5-6, pg 5-64, NEDO-24011)

Thermal Power (% of rated)	
BOC to EOC-1000 MWd/t	100
EOC-1000 MWd/t to EOC	99
Turbine Pressure (psig)	950
GETAB Initial Conditions	
Reactor Core Pressure (psia)	1035
Inlet Enthalpy (Btu/lb) <i>370'</i>	523.7

## LOADING ERROR

Linear Heat Generation Rate (kW/ft)	16.9
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ENCLOSURE III

QUAD-CITIES UNIT 1

NRC DOCKET NO. 50-254

NEDO-24147  
78NED285  
Class I  
September 1978

QUAD CITIES NUCLEAR POWER STATION  
UNIT 1 RELOAD 4  
SUPPLEMENTAL LICENSING INFORMATION  
FOR  
BARRIER LEAD TEST ASSEMBLIES

---

NUCLEAR ENERGY PROJECTS DIVISION • GENERAL ELECTRIC COMPANY  
SAN JOSE, CALIFORNIA 95125

GENERAL  ELECTRIC

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## 1. INTRODUCTION

This document provides the supplemental information for four 8x8 Barrier Lead Test Assemblies (BLTA) which are part of Reload No. 4, Cycle 5, at the Quad Cities 1 Nuclear Power Station. The generic design information and safety analyses for standard 8x8R fuel given in References 1 and 2 are applicable to the Barrier Lead Test Assemblies except as noted in the following supplemental information.

### 1.1 OBJECTIVES

The Barrier Lead Test Assembly Program is part of a larger demonstration program, which is intended to provide early experience in a commercial power reactor with fuel rod designs developed for their potential capability to remedy the pellet-cladding interaction (PCI) fuel rod failure mechanism. The primary objective of the Barrier Lead Test Assembly Program is to accumulate burnup ahead of a large-scale demonstration to provide assurance that, while the remedy resists PCI, it is not subject to some unforeseen problem that becomes manifest at high burnup. Further objectives of the program are to help define manufacturing process parameters and provide a source of prototypical lead burnup fuel rods which would be available for testing or destructive examination.

### 1.2 SCOPE

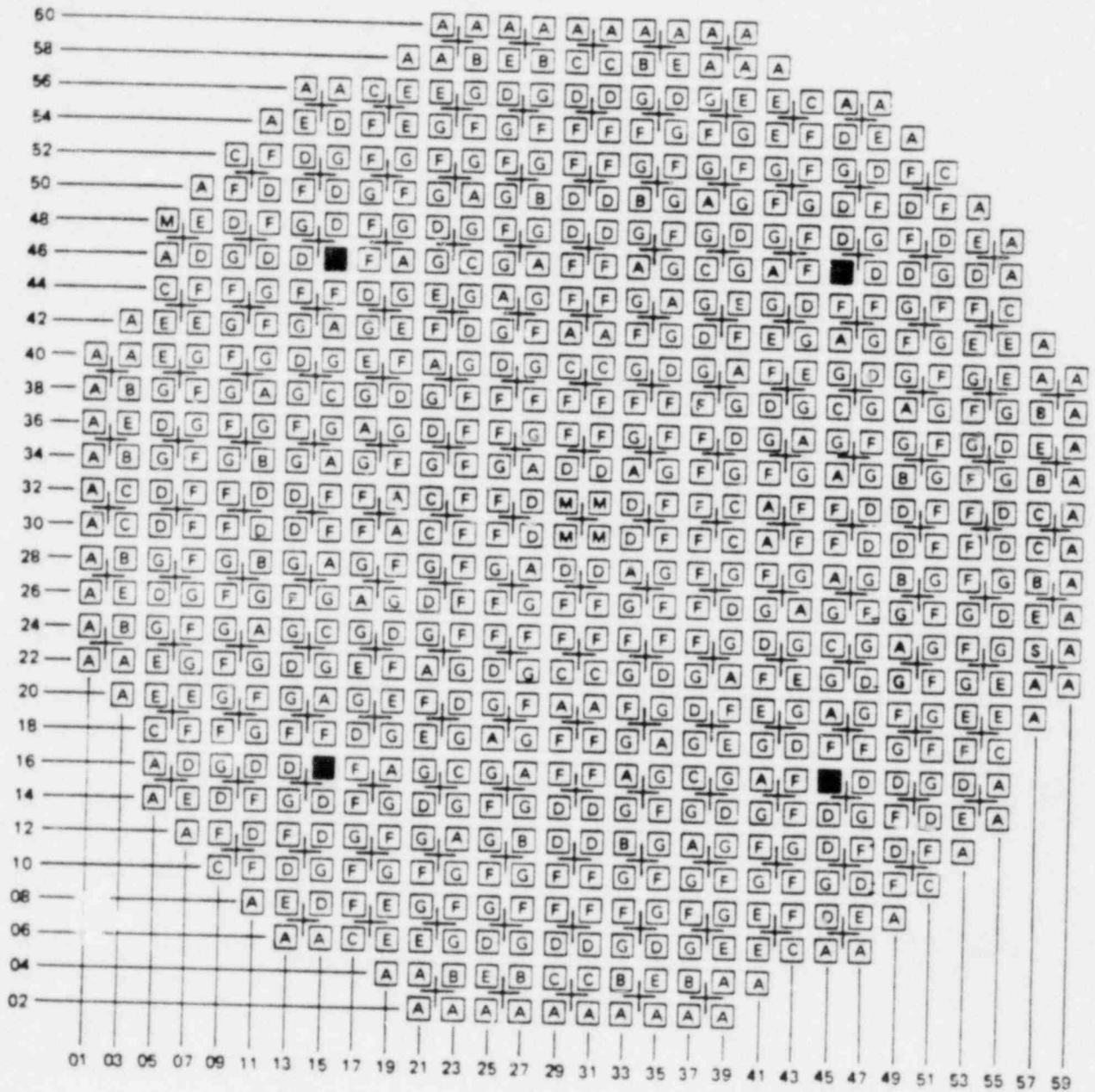
The four Barrier Lead Test Assemblies to be loaded as part of Reload 4 at Quad Cities 1 are targeted to be operated for at least four full reactor cycles. The four bundles are all similar in design (Section 3.1) and will be located in symmetric core locations to obtain similar operating histories. Location of the BLTA's in the core has been selected such as to assure relatively high power operation and to minimize the impact of the BLTA's on the operation of Quad Cities 1.

The performance of these four BLTA's will be closely monitored by the General Electric Company through the following tasks: (1) specific pre-irradiation characterization measurements will be performed; (2) archive samples will be retained; and (3) an inspection program during each reactor outage will be implemented subject to Commonwealth Edison Company approval and the availability of authorized funds. One of the tasks may also include the replacement of selected fuel rod segments. The removed segments would be available for testing or destructive examination, thus providing additional performance data. A detailed post-irradiation examination of the BLTA full-length fuel rods could be performed if the interim inspection program indicates that such an examination would be beneficial.

## 2. SUMMARY

The Barrier Lead Test Assemblies are the same 8x8 lattice configuration and have the same dimensions as General Electric Company's standard 8x8 retrofit fuel (Reference 1). Notable BLTA mechanical differences with the 8x8 retrofit fuel (Table 3-1) are the use of two segmented rods, a cladding inside surface which is either lined with high purity zirconium or plated with copper in all fuel rods, and fuel rod prepressurization of three atmospheres. Enrichments are the same as the 8DRB265L bundle. The segmented rods use the same enrichment as the rods they replace, but contain no natural  $UO_2$ . The BLTA has been evaluated with specific attention to the noted mechanical differences, and results show that all design requirements are satisfied.

Core locations of the BLTA's are shown in Figure 2-1. Evaluations show that the BLTA's, as located, will not limit operation of the core. Safety analyses indicate that there is an insignificant effect on the Quad Cities 1 core characteristics resulting from loading the four BLTA's in the locations shown.



FUEL TYPE	
A = INITIAL FUEL	D = RELOAD 2 (80B250)
B = RELOAD 1 (TDB230) (M = MIXED OXIDE) (S = SEGMENTED TEST ROD)	E = RELOAD 2 (80B262)
C = RELOAD 1 (80B250)	F = RELOAD 3 (80B250)
	G = RELOAD 4 (80RB265L)
	■ = BARRIER LTA

Figure 2-1. Quad Cities Unit 1 Reload 4 Design Reference Core Loading Showing BLTA Locations



### 3. MECHANICAL DESIGN

#### 3.1 GENERAL DESCRIPTION

##### 3.1.1 Bundle

The Barrier Lead Test Assembly (BLTA) design is structurally the same as the 8x8 retrofit (8x8R) design which is being applied to Quad Cities 1 Reload 4. A limited number of bundle modifications have been made to accommodate the barrier cladding on all fuel rods and to provide short fuel rod segments which could be removed for later testing. This section reviews the BLTA design and provides a detailed description of the bundle changes. Table 3-1 lists significant design parameters of the BLTA and provides a direct comparison with the 8DRB265L reload fuel bundle.

The BLTA fuel bundle contains 60 full-length fuel rods, two segmented fuel rods of four segments each, and two water rods, one of which is also a spacer positioning rod. The rods are spaced and supported in a square (8x8) array by the upper and lower tieplates in the same manner as shown on Figure 2-1 of Reference 1. The lower tieplate has a nosepiece which supports the fuel assembly in the reactor. The upper tieplate has a handle for transferring the fuel bundle from one location to another. The identifying assembly serial number is engraved on the top of the handle, and a boss projects from one side of the handle to aid in ensuring proper fuel assembly orientation. Both upper and lower tieplates are fabricated from Type-304 stainless steel castings. Zircaloy-4 fuel rod spacers equipped with Inconel-X springs are employed to maintain rod-to-rod spacing. Finger springs are also employed with the BLTA design at the lower tieplate to channel interface. The BLTA fuel assembly outline dimensions are the same as those of the 8DRB265L reload fuel to be used for Quad Cities 1 Reload 4. The BLTA uses identical tieplates, spacers and finger springs as the 8DRB265L fuel bundle.

### 3.1.2 Fuel Rods

Three types of fuel rods are used in the BLTA. In addition to tie rods and standard rods, there are two segmented rods, consisting of four segments each, which replace two of the standard rods in the 8DRB265L fuel bundle. Each fuel rod consists of high-density ceramic uranium dioxide fuel pellets stacked within a barrier cladding which is evacuated, backfilled with helium, and sealed with Zircaloy end plugs welded in each end. The helium backfill pressure is 3 atm at room temperature. Recent studies have shown that a larger inventory of helium gas (achieved by using a higher cold internal pressure) improves the gap conductance between fuel pellets and cladding resulting in reductions in fuel temperature, thermal expansion and fission gas release. The pressurized rods operate at effectively lower linear heat generation rates (kW/ft) and are, therefore, expected to yield performance benefit in terms of increased fuel reliability. The effect of 3 atm prepressurization in the BLTA's is negligible on Quad Cities 1 core performance and plant operation. The prepressurization (3 atm) selected in this design also results in improved margin to MAPLHGR limits by reducing stored energy. Experience with prepressurized fuel rods has been obtained in General Electric's ongoing Pressurized Test Assembly (PTA) Program (Section 3.3).

The eight fueled tie rods in each bundle have threaded end plugs which thread into the lower tieplate casting and extend through the upper tieplate casting. A stainless steel hexagonal nut and locking tab are installed on the upper end plug to hold the assembly together. These tie rods support the weight of the assembly during fuel-handling operations only when the assembly hangs by the handle; during operation, the fuel rods are supported by the lower tieplate. Except for the inside surface of the cladding and the 3 atm prepressurization, the tie rods are identical to those used in the 8DRB265L fuel bundles. Standard nuts and locking tabs are used for their assembly.

Each of the two segmented rods in each bundle consists of four identical fuel rod segments, an upper extension plug, and a lower extension plug. Each segment has a threaded upper end plug and a threaded hole in its lower end plug, which make them completely interchangeable. The extension plugs are designed to provide the same interface with the tieplates as the end plugs on standard full-length fuel rods. A complete segmented rod is assembled by

screwing the extension plugs and four identical segments together. The positions of the segmented rods in the BLTA were selected to maintain the diagonal symmetry of the bundle and to minimize the impact of potential flux peaking in adjacent fuel rods. Experience with similar segmented fuel rods has been obtained in three specially designed Segmented Test Rod bundles which have been in operation since 1974 (see Section 3.3).

The remaining 52 fuel rods in a bundle are basic fuel rods having the same active fuel length as the tie rods. The end plugs of the standard rods have pins which fit into anchor holes in the tieplates. An Inconel-X expansion spring located over the top end plug pin of each fuel rod keeps the fuel rods seated in the lower tieplate and allows them to expand axially and independently by sliding within the holes of the upper tieplate. Except for the barrier or liner on the inside surface of the cladding and the 3 atm prepressurization, the BLTA basic fuel rods are identical to the standard 8DRB265L fuel rods. Standard expansion springs are used with the BLTA rods.

Adequate free volume is provided within each fuel rod and fuel rod segment in the form of a pellet-to-cladding gap and a plenum region at the top of the fuel rod to accommodate thermal and irradiation expansion of the  $UO_2$  and the internal pressures resulting from the 3 atm helium fill gas, impurities, and gaseous fission products liberated over the design life of the fuel. A plenum spring, or retainer, is provided in the plenum space to minimize movement of the fuel column inside the fuel rod during fuel shipping and handling. A hydrogen getter is also provided in the plenum space as assurance against chemical attack from the inadvertent admission of moisture or hydrogenous impurities into a fuel rod during manufacturing. Standard retainers and getters are used in all full-length rods, and similar configurations are applied to the segmented rod, each of which has its own independent plenum.

### 3.1.3 Water Rods

The two water rods used in the BLTA are identical to the 8x8R fuel bundle water rods, and their functions are the same. One of these is used to position seven Zircaloy-4 spacers. The spacer-positioning water rod is assembled to the spacers by sliding the rod through the spacer cell. The rod is then rotated so that the tabs bracket the elements of the spacer structure, thereby positioning the spacer in the required axial position. The rod is prevented from rotating to unlock the spacers by the engagement of its (square) lower end plug with

a square tieplate hole. Several holes are drilled around the circumference of the water rods near each end to allow coolant water to flow through the rods.

#### 3.1.4 Fuel

The BLTA uses the same fuel pellets as the 8DRB265L fuel bundle. The fuel consists of high-density ceramic uranium dioxide, manufactured by compacting and sintering uranium dioxide powder into cylindrical pellets with chamfered edges. The average  $UO_2$  pellet immersion density is approximately 95% of theoretical density. The pellet dimensions are given in Table 3-1.

Eight different U-235 enrichments are used in the fuel assemblies to reduce the local power peaking factor (Figure 3-1). Fuel element design and manufacturing procedures have been developed to prevent errors in enrichment location with a fuel assembly. The fuel rods within each assembly are designed with characteristic mechanical end fittings and are marked with an enrichment identification for each enrichment.

The BLTA bundle incorporates the use of small amounts of gadolinium as a burnable poison in selected fuel rods. The gadolinia-uranium fuel rods are designed with characteristic extended end plugs, which are the same as used on the 8DRB265L fuel bundle gadolinia-uranium rods. These extended end plugs permit a positive, visual check on the location of each gadolinium-bearing rod after bundle assembly.

Figure 3-1 shows the location of the various fuel rod types within the Barrier Lead Test Assemblies. With the exception of two segmented rods and either a zirconium liner or a copper barrier on the inside surface of the fuel rod cladding, the BLTA rod types and locations are the same as the 8DRB265L reload bundle. Axial locations of the natural uranium in the fuel rods are shown in Figure 3-2. The BLTA fuel rods contain the same enrichments and gadolinia loading as corresponding rods in the 8DRB265L bundle. The BLTA segmented rods also contain the same enrichment as corresponding rods but the total active fuel length is shorter and no natural  $UO_2$  is used. At both ends of every segmented rod fuel column there are 0.5-in. hafnia-yttria pellets, which are placed there to assure that no adverse peaking will occur in the bundle due to these segments. These flux depressor pellets are used in all three operating Segmented Test Rod bundles (Section 3.3).

### 3.1.5 Cladding

Overall dimensions of the BLTA fuel rod cladding are identical to the Quad Cities 1 Reload 4 fuel rod cladding. The thin copper barrier or zirconium liner on the inside surface displaces the same thickness of Zircaloy-2 material to maintain the same total cladding thickness. The BLTA tubing with this slight reduction in Zircaloy-2 material is still adequate to be essentially free standing in the 1000 psi BWR external pressure environment.

The following variations of barrier or liner cladding will be used (one BLTA of each type):

- (1) Cu-barrier (0.4 mil)  
Electroless Cu-plating on Zr-2 cladding
- (2) Cu-barrier (0.4 mil)  
Electroless Cu-plating on autoclaved (oxidized) Zr-2 cladding
- (3) Zr-liner (3.0 mil)  
Crystal bar zirconium co-extruded with the Zr-2 cladding
- (4) Zr-liner (3.0 mil)  
Low oxygen sponge zirconium co-extruded with the Zr-2 cladding

The copper is plated on the inner surface of all the Zircaloy-2 cladding by flowing a series of solutions through the tubes which first prepare and then electrolessly plate the inside surface. The procedure for plating copper directly onto the Zircaloy-2 cladding starts with annealed tubings after a chemical polish (pickle) and prior to the normal autoclave step. A similar procedure is used for plating on the autoclaved tubes except that the plating is done after some initial surface preparation and the standard autoclave. All the completed cladding has the same autoclaved outside surface as standard 8x8R fuel rod cladding.

Rigid inspection techniques, including 100% ultrasonic testing for flaws, are applied to the tubes prior to plating. After plating, the tubes are nondestructively examined to measure copper thickness, and representative samples are destructively tested to assure good adherence of the copper to the Zircaloy-2.

The high purity zirconium liner tubes are fabricated by co-extrusion of either the crystal bar or low oxygen sponge zirconium with the Zircaloy-2 cladding. Extrusion billets of Zircaloy-2 are machined to accept liners of high purity zirconium as close-fitting sleeves inside them. Extrusion of these composite billets is then performed in a normal manner using standard lubricants and operating parameters. This process produces a high quality tubing, which has a uniform liner thickness and an excellent mechanical bond with the Zircaloy-2. As a final process step, the outside surface of the zirconium liner tubes is also autoclaved the same as the production 8x8R tubes.

The same rigid inspection techniques are applied after processing the zirconium liner tubes. This includes tight dimensional control and ultrasonic flaw detection.

### 3.1.5 Materials

All of the BLTA materials which are exposed to the reactor coolant environment are identical to the 8x8R reload fuel. One new material, copper, is introduced in a 0.4 mil layer on the inside surface of all rods in two of the BLTA's. Fuel rods in the other two test assemblies have a 3-mil inside layer of high purity zirconium (either crystal bar or low oxygen sponge material), which is also the principal constituent in the Zircaloy-2 cladding.

#### 3.1.6.1 Copper

The copper barrier is at least 99% pure copper and forms an adherent plating to the Zircaloy-2 cladding. This very thin copper barrier has been disregarded in the mechanical design analysis because of its negligible structural contribution.

The benefit from copper is derived from its ability to protect the Zircaloy-2 inside surface from fission products which could potentially contribute to pellet-cladding interaction (PCI) failures. Tests conducted to date (Reference 4) have confirmed this ability, and also have thus demonstrated the compatibility of copper with both  $UO_2$  fuel and Zircaloy-2 cladding.

### 3.1.6.2 Zirconium

The zirconium liner (either the crystal bar or low oxygen sponge material) is high purity zirconium, which is coextruded with the Zircaloy-2 to form an excellent mechanically bonded cladding. This composite cladding meets the same tensile strength requirements as standard 8x8R cladding. However, the detailed mechanical analysis of the cladding was performed using the actual lower zirconium strength properties for the applicable liner region. This resulted in an effectively thinner cladding than the standard product from a mechanical standpoint.

Values of thermal conductivity, coefficient of thermal expansion, and elongation of Zircaloy-2 were used for the zirconium liner material.

The benefit of the zirconium liner also is derived from its ability to resist PCI failures. Extensive tests to date on this cladding configuration (Reference 4) have confirmed its PCI benefit as well as the material compatibility of zirconium in the fuel rod environment.

## 3.2 THERMAL AND MECHANICAL EVALUATIONS

The same safety and design evaluations as performed for standard 8x8R fuel (Reference 1, Sections 2.4 and 2.5) have been applied to the BLTA designs. The models used for these evaluations are also the same as described in Reference 1. Because the BLTA's use the same bundle components and the same fuel rod dimensions as the 8x8R fuel bundle, most of the fuel assembly mechanical analyses are directly applicable to the BLTA's. Effects due to the unique BLTA cladding, prepressurization and segmented rods have been accounted for in the BLTA safety and design evaluations.

### 3.2.1 Results from Safety Evaluation

The calculated LHGR resulting in 1% plastic strain in the cladding is equal to or greater than 160% of the design maximum steady-state power throughout life for all rod types in the BLTA. This ratio considers the presence of a calculated power spiking allowance for densification being added to the maximum LHGR.

### 3.2.2 Results from Design Evaluations

#### 3.2.2.1 Steady-State Mechanical Performance

BLTA fuel is designed to operate at core rated power with sufficient design margin to accommodate reactor operations and satisfy the mechanical design applied to 8x8R fuel. In order to accomplish this objective, the BLTA fuel is designed to operate at a maximum steady-state linear heat generation rate (LHGR) of 13.4 kW/ft, plus an allowance for densification power spiking.

Thermal and mechanical analyses demonstrate that the mechanical design criteria are met for the maximum operating power and exposure combination throughout the BLTA fuel life. Design analyses performed for the BLTA fuel show that the stress intensity limits given in Table 2-6 of Reference 1 are not exceeded during continuous operation with LHGR's up to the operating limit of 13.4 kW/ft, nor for short-term transient operation up to 16% above the peak operating limit of 13.4 kW/ft (i.e., 15.6 kW/ft), plus an allowance for densification power spiking. Stresses due to external coolant pressure, internal gas pressure, thermal gradients, spacer contact, flow-induced vibration, and manufacturing tolerances were considered. The maximum internal pressure is applied coincident with the minimum applicable coolant pressure.

#### 3.2.2.2 Fatigue Analysis

During fuel life, less than 15% of the allowable fatigue life of the BLTA fuel is consumed.

#### 3.2.2.3 Incipient Fuel Center Melting

The LHGR's required to reach center melting in BLTA fuel rods are at least as high as the values shown in Table 2-4 of Reference 1. The higher gap conductance due to prepressurization raises the required LHGR's for center melting, while the effect of the copper barrier or zirconium liner is negligible.



### 3.2.2.4 Fuel Cladding Temperatures

BLTA fuel cladding temperatures, as a function of heat flux, are very similar to the standard 8x8R cladding temperatures shown in Reference 1, Figure 2-13 for late-in-life conditions. The cladding geometries are the same and the effect of the barrier or liner is negligible.

### 3.2.2.5 Densification Analysis

#### 3.2.2.5.1 Power Spiking Analysis

The equation employed to calculate maximum gap size is described in Reference 3. The BLTA active fuel length and use of natural uranium at the ends of the fuel column are the same as the Quad Cities 1 8x8R reload fuel and have the same effect on maximum gap size. The power increase as a function of axial position (as described in paragraph 2.4.2.1.1 of Reference 1) has been added to the BLTA license limit LHGR (13.4 kW/ft) and considered in the BLTA design and safety analyses, wherever applicable.

#### 3.2.2.5.2 Cladding Creep Collapse

A creep collapse analysis of BLTA fuel was performed using the same bases as described in paragraph 2.5.3.1.1 of Reference 1. The internal pressure due to helium backfill during fabrication was considered. The higher backfill pressure (3 atm) is beneficial to the BLTA compared to the 8x8R design. No credit was taken for internal gas pressure due to released fission gas or volatiles. A thinner cladding was assumed for the BLTA creep-collapse analysis to conservatively account for the lower strength of the Zr liner. Based on the analysis results, cladding collapse was not calculated to occur for the BLTA design.

#### 3.2.2.5.3 Increased Linear Heat Generation Rate

BLTA design changes have no effect on the conclusion for 8x8R fuel (Reference 1) that the pellet decrease in length due to densification is less than the increase in length due to thermal expansion.

#### 3.2.2.5.4 Stored Energy Determination

The effects on stored energy due to densification are accounted for in the LOCA evaluation.

### 3.3 FUEL OPERATING AND DEVELOPMENTAL EXPERIENCE

The basic structure of the Barrier Lead Test Assemblies and the configuration of their fuel rods are the same as the standard reload fuel for Quad Cities 1 Reload 4 (Reference 2). The unique BLTA features:

- (1) copper barrier or zirconium liner cladding,
- (2) segmented rods, and
- (3) prepressurization

have all been included in other fuel assemblies which are currently in operation and which have been extensively evaluated through both analysis and testing.

The program under which the BLTA's are being administered also includes a task that provides operating experience and other supporting test data (Reference 4). Power reactor experience with fuel having the above features has been obtained through the use of three Segmented Test Rod bundles which have been operating since 1974. More than 350 segments, of the same basic design as the 8 segments in each BLTA, have been irradiated in these three test bundles. The segments contain a variety of different fuel design concepts, including both copper barrier cladding and zirconium liner cladding. Wall thicknesses of the barrier segments have ranged from 28 mils to 34 mils (BLTA rods are 32 mils thick) and prepressurization has ranged from 1 atm to 17 atm of helium. All recent replacement barrier segments have been prepressurized to 3 atm. A complete summary of the status of all copper barrier and zirconium liner segments may be found in Reference 4.

Several Cu-Barrier and Zr-Liner segments have been removed from the Segmented Test Rod bundles, and eight of these segments have been ramp tested to 18 kW/ft or higher in the General Electric Test Reactor (GETR). The first two segments tested had burnups between 4000 and 5000 MWd/t; the remaining segments ranged from 7200 to 9200 MWd/t when ramp tested. None of the segments experienced

a pellet-cladding interaction failure. One of the early tests of a copper barrier fuel rod did result in failure caused by substantial center melting, which strained the cladding beyond its yield point. The other copper barrier segment in the first test series also experienced extensive fuel center melting but survived the ramp test. This experience base will continue to be expanded upon as the Segmented Test Rod Program proceeds, and will continue to provide information supporting BLTA operation. In addition to this power reactor operating experience with segmented rods, a number of other supporting tests are also described in Reference 4.

Prepressurization of General Electric BWR fuel rods with helium to 3 atm has been extensively studied in preparation for a design change, which is now planned to be implemented during 1979 (Reference 5). Part of this preparation has included the operation of a Pressurized Test Assembly (PTA) in Peach Bottom Unit 3 since April 1977. This fuel assembly has the same 8x8R fuel rod geometry as the BLTA's and contains 24 rods that are prepressurized to 3 atm (Reference 6). A visual examination of the assembly was performed at the Peach Bottom site in April 1978. The mechanical integrity of the PTA was confirmed with no abnormal conditions observed. Further interim examinations of the PTA will be performed contingent on the availability of the fuel as influenced by plant operation.

Table 3-1  
8DRB265 RELOAD FUEL ASSEMBLY AND BLTA DESIGN PARAMETERS

	8DRB265L	BLTA
<b>Fuel Assembly</b>		
Geometry	8x8	8x8
Rod Pitch (in.)	0.64	0.64
Water to Fuel Volume Ratio	2.75	2.75
Heat Transfer Area (ft <sup>2</sup> )	94.9	94.3
Weight of UO <sub>2</sub> (Kg)	200.5	199.3
Weight of U (Kg)	176.8	175.7
Average Enrichment (w/o U-235)	2.65	2.65
Finger Springs	Yes	Yes
<b>Full-Length Fuel Rods</b>		
Number	62	60
*Active Fuel Length (in.)	145.24	145.24
Gas Plenum Length (in.)	9.48	9.48
Fill Gas	Helium	Helium
Fill Gas Pressure (atm)	1	3
<b>Fuel Rod Segments</b>		
Number	0	8
Active Fuel Length (in.)	N/A	29.63
Gas Plenum Length (in.)	N/A	5.5
Fill Gas	N/A	Helium
Fill Gas Pressure (atm)	N/A	3
<b>Fuel</b>		
Material	Sintered UO <sub>2</sub>	Sintered UO <sub>2</sub>
Pellet Diameter (in.)	0.410	0.410
Pellet Length (in.)	0.410	0.410
Pellet Immersion Density (% TD)	95.0	95.0
<b>Cladding</b>		
Material	Zr-2	Zr-2/Cu or Zr-2/Zr
Outside Diameter (in.)	0.483	0.483
Thickness (Including Barrier or Liner, if any)	0.032	0.032

Table 3-1

## 8DRB265 RELOAD FUEL ASSEMBLY AND BLTA DESIGN PARAMETERS (Continued)

	8DRB265L	BLTA
Copper Plating Thickness	N/A	0.0004 in.
or		
Zirconium Liner Thickness	N/A	0.003 in.
<b>Water Rod</b>		
Material	Zr-2	Zr-2
Outside Diameter (in.)	0.591	0.591
Thickness	0.030	0.030
<b>Spacers</b>		
Material	Zr-4 with Inconel	Zr-4 with Inconel
Number per bundle	X-750 Springs 7	X-750 Springs 7
<b>Fuel Channel</b>		
Material	Zr-4	Zr-4
Outside Dimension (in.)	5.438	5.438
Wall Thickness (in.)	0.080	0.080

\*Includes natural Uranium (12 in. per full-length rod).

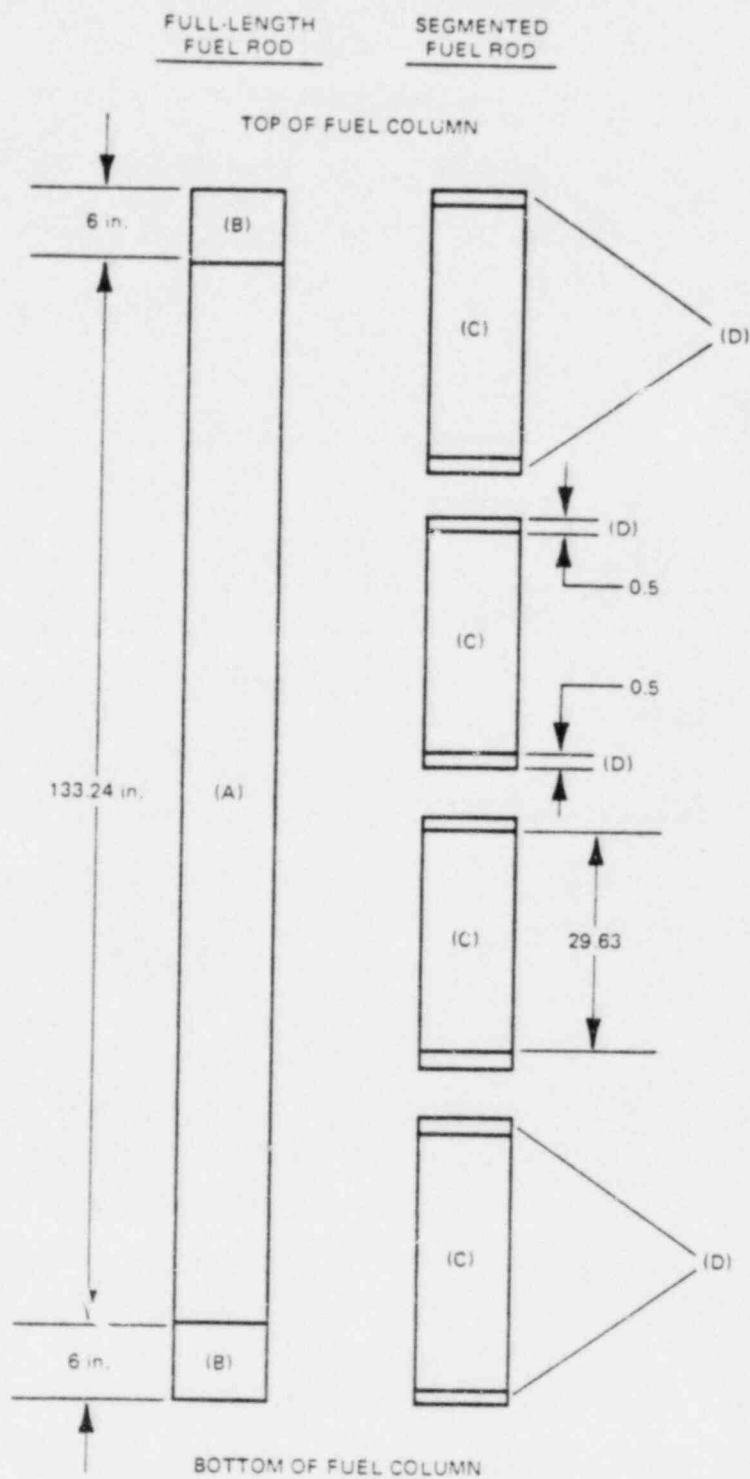
BLADE

CONTROL	6	5	4	4	4	4	4	5
	5	4	2	Gd	2	2	2	4
	4	2	2	2	1	1	Gd	2
	4	Gd	2	2S	H <sub>2</sub> O	1	1	2
	4	2	1	H <sub>2</sub> O	1	1	Gd	2
	4	2	1	1	1	1	1	2
	4	2	Gd	1	Gd	1	2S	3
	5	4	2	2	2	2	3	4

ROD TYPE	U-235 wt%	NUMBER OF RODS
1	3.8	14
2	3.0	19
3	2.4	2
4	2.0	14
5	1.7	4
6	1.3	1
Gd	3.0	6
H <sub>2</sub> O	3.0	2
2S (SEGMENTED)	3.0	2
		64

\*SPACER POSITIONING

Figure 3-1. Barrier Lead Test Assembly Lattice



- (A) 2.82%, AVERAGE IN ALL RODS.
- (B) 0.71%, ALL RODS EXCEPT 2 SEGMENTED RODS
- (C) 3.0%, ALL SEGMENTS
- (D) HAFNIA-YTTRIA PELLETS.

Figure 3-2. Barrier Lead Test Assembly U-235 Enrichment Axial Profile

4. THERMAL-HYDRAULIC ANALYSIS

Discussion of steady-state hydraulic models is presented in Section 4 of Reference 1. The BLTA is treated the same as an 8x8R bundle in the Quad Cities 1, Cycle 5 thermal-hydraulic analysis. Pressure drops through BLTA and standard 8x8R bundles are considered to be the same because identical bundle components and the same fuel rod geometries are used. Relative pressure drop effects due to the two segmented rods in each BLTA are negligible.



## 5. NUCLEAR CHARACTERISTICS

The zirconium liners have no impact on the BLTA nuclear performance. The copper barriers do not affect bundle rod-to-rod power distributions, but do cause a small (less than 0.2%) decrease in bundle reactivity.

In the nonfueled regions of the BLTA's, the segmented rods cause local increases in the maximum peak-to-average rod powers. However, this local effect is counteracted by decreases in total bundle power in the same regions. In the natural uranium regions of the bundles, the enriched fuel of the segmented rods experience large increases in peak-to-average rod powers. However, since these are low bundle power regions, the local increases are not significant.

In summary, nuclear analyses have shown that the effects of the two segmented rods and either the Zr-liner or the Cu-barrier cladding in each BLTA will have no significant effect on the nuclear performance of the BLTA's relative to the 8DRB265 reload fuel in Quad Cities 1.

## 6. REACTOR LIMITS DETERMINATION

### 6.1 GETAB LIMITS

#### 6.1.1 GETAB Transient Results

The most severe transient for a Barrier Lead Test Assembly from rated conditions is a load rejection with failure of the bypass valves. This event results in a maximum  $\Delta$ CPR of 0.25. Addition of this  $\Delta$ CPR to the safety limit MCPR gives the minimum initial MCPR to avoid violating the safety limit MCPR during the most severe transient involving a BLTA.

#### 6.1.2 MCPR Operating Limit

Based on a safety limit MCPR of 1.07 for this cycle, the operating limit for the Barrier Lead Test Assemblies is 1.32. This is 0.03 higher than the 8x8R fuel assemblies because of a conservative treatment of the effects of pre-pressurization on transient performance of the BLTA design. A more comprehensive analysis comparing 3 atm and 1 atm initial pressures has shown an insignificant (0.01) difference in  $\Delta$ CPR (Reference 5, Section 4.2.2).

#### 6.1.3 Transient Analysis Initial Condition Parameters

The values used as initial input conditions for the BLTA transient analysis are shown in Table 6-1.

### 6.2 STABILITY ANALYSIS

The addition of four Barrier Lead Test Assemblies to the Quad Cities 1 core will have a negligible impact on the reactor core stability and the channel hydrodynamic performance as compared to 8DRB265L reload fuel bundles (Reference 2, Section 13).

### 6.3 ACCIDENT EVALUATIONS

#### 6.3.1 Loss-of-Coolant Accident Results

An emergency core cooling system (ECCS) analysis has shown that the effects of a loss-of-coolant accident on the Barrier Lead Test Assemblies results in a slight increase in MAPLHGR and lower peak cladding temperatures (PCT) compared to the 8x8R reload bundle. This improvement is caused by the higher initial pressurization of the BLTA, which increases the gap conductance and reduces the stored energy in the fuel rods.

Results of the ECCS analysis are presented in Table 6-2.

The potential effect that the barriers could have on the overall cladding strength and, consequently, the calculated PCT, was considered by running an additional analysis which neglected the structural strength supplied by the 3-mil zirconium liner. Results of this separate analysis indicated the potential for a maximum of 10 fuel rod perforations at an exposure of 30,000 Mwd/t for the highest power bundle. No perforations were predicted for the copper barrier fuel where no significant reduction in cladding strength is expected to occur due to the thinness of the copper plating.

#### 6.3.2 Loading Error Accident

The analysis and results of a loading error involving either a rotated BLTA or a misplaced BLTA in an 8x8 or 7x7 location in Quad Cities 1, Cycle 5 are conservatively represented by the SDRB265L reload bundle loading error results (Reference 2, Section 15).

#### 6.3.3 Control Rod Drop Accident

Since the nuclear characteristics of the BLTA are nearly identical to the standard reload bundle, the analysis and results of the control rod drop accident for the reload (Reference 2, Section 16) is considered applicable to the BLTA fuel.

Table 6-1  
GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS

Peaking Factors (local, radial, axial)	BLTA
R-Factor	(1.22, 1.79, 1.40)
Bundle Power, MWt	1.051
Bundle Flow, $10^3$ lb/hr	6.082
Initial MCPR	108.57
	1.32

Table 6-2  
BARRIER LEAD TEST ASSEMBLY  
MAPLHGR, PCT, OXIDATION FRACTION VERSUS EXPOSURE

EXPOSURE (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	OXIDATION FRACTION
200	11.6	2171	0.032
1,000	11.6	2178	0.033
5,000	12.0	2198	0.034
10,000	12.1	2195	0.033
15,000	12.0	2200	0.033
20,000	11.7	2187	0.032
25,000	11.0	2115	0.056
30,000	10.5	2029	0.042

7. REFERENCES

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ENCLOSURE IV

QUAD-CITIES UNIT 1

NRC DOCKET NO. 50-254

NEDO-24146  
78NED284  
Class I  
September 1978

LOSS-OF-COOLANT ACCIDENT ANALYSIS REPORT

FOR

DRESDEN UNITS 2, 3 AND QUAD CITIES UNITS 1, 2  
NUCLEAR POWER STATIONS

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NUCLEAR ENERGY PROJECTS DIVISION • GENERAL ELECTRIC COMPANY  
SAN JOSE, CALIFORNIA 95125

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## 1. INTRODUCTION

The purpose of this document is to provide the results of the loss-of-coolant accident (LOCA) analysis for the Dresden Units 2, 3 and Quad Cities Units 1, 2 Nuclear Power Stations (D2,3/QC1,2) with a partial core loading of reload fuel with holes drilled in the lower tieplates. The analysis was performed using approved General Electric (GE) calculational models.

This reanalysis of the plant LOCA is provided in accordance with the NRC requirement (Reference 1) and to demonstrate conformance with the ECCS acceptance criteria of 10CFR50.46. The objective of the LOCA analysis contained herein is to provide assurance that the most limiting break size, break location, and single failure combination has been considered for the plant. The required documentation for demonstrating that these objectives have been satisfied is given in Reference 2. The documentation contained in this report is intended to satisfy these requirements.

The general description of the LOCA evaluation models is contained in Reference 3. Recently approved model changes (Reference 4) are described in References 5 and 6. These model changes are employed in the new REFLOOD and CHASTE computer codes which have been used in this analysis. In addition, a model which takes into account the effects of drilling alternate flow path holes in the lower tieplate of the fuel bundle and the use of such fuel bundles in a full or partial core loading is described in References 7, 8, and 9. This model was also approved in Reference 4. Also included in the reanalysis are current values for input parameters based on the LOCA analysis reverification program being carried out by GE. The specific changes as applied to D2,3/QC1,2 (partial drill) are discussed in more detail in later sections of this document.

Plants are separated into groups for the purpose of LOCA analysis (Reference 10). Within each plant group there will be a single lead plant analysis which provides the basis for the selection of the most limiting break size yielding the highest peak cladding temperature (PCT). Also, the lead plant analysis provides an expanded documentation base to provide added insight into evaluation of the details of particular phenomena. The remainder of

the plants in that group will have non-lead plant analyses referenced to the lead plant analysis. This document contains the non-lead plant analysis for D2,3/QC1,2 which are now BWR/3's in the BWR/4 with loop selection logic group of plants and is consistent with the requirements outlined in Reference 2.

The same models and computer codes are used to evaluate all plants. Changes to these models will cause changes in phenomenological responses that are similar within any given plant group. The difference in input parameters are not expected to result in significantly different results for the plants within a given group. Emergency core cooling system (ECCS) and geometric differences between plant groups may result in different responses for different groups but within any group the responses will be similar. Thus, the lead plant concept is still valid for this evaluation.

2. LEAD PLANT SELECTION

Lead plants are selected and analyzed in detail to permit a more comprehensive review and eliminate unnecessary calculations. This constitutes a generic analysis for each plant of that type which can be referenced in subsequent plant submittals.

The lead plant for D2,3/QC1,2 with drilled fuel is Duane Arnold. The justification for categorizing D2,3/QC1,2 in this group of plants is the same as for Pilgrim and the lead plant analysis for this group is presented in Reference 11.

3. INPUT TO ANALYSIS

A list of the significant plant input parameters to the LOCA analysis is presented in Table 1.

Table 1  
SIGNIFICANT INPUT PARAMETERS TO THE  
LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters

Core Thermal Power	2578 MWt, which corresponds to 102% of rated power
Vessel Steam Output	$9.96 \times 10^6$ lbm/h, which corresponds to 102% of rated power
Vessel Steam Dome Pressure	1020 psia
Recirculation Line Break Area for Large Breaks - Suction	4.18 ft <sup>2</sup> (DBA)
Number of Drilled Bundles	156

Fuel Parameters:

<u>Fuel Type</u>	<u>Fuel Bundle Geometry</u>	<u>Peak Technical Specification Linear Heat Generation Rate (kW/ft)</u>	<u>Design Axial Peaking Factor</u>	<u>Initial Minimum Critical Power Ratio*</u>
A. 7D212 - No Gad	7x7	17.5	1.57	1.2
B. 7D212L	7x7	17.5	1.57	1.2
C. 7D230	7x7	17.5	1.57	1.2
D. EEIC - Pu	7x7	17.5	1.57	1.2
E. 8D250	8x8	13.4	1.57	1.2
F. 8D262	8x8	13.4	1.57	1.2
G. SDRB265L	8x8	13.4	1.57	1.2
H. Barrier LTA	8x8	13.4	1.57	1.2

\* To account for the 2% uncertainty in bundle power required by Appendix K, the SCAT calculation is performed with an MCPR of 1.18 (i.e., 1.2 divided by 1.02) for a bundle with an initial MCPR of 1.20.

#### 4. LOCA ANALYSIS COMPUTER CODES

##### 4.1 RESULTS OF THE LAMB ANALYSIS

This code is used to analyze the short-term blowdown phenomena for large postulated pipe breaks (breaks in which nucleate boiling is lost before the water level drops and uncovers the active fuel) in jet pump reactors. The LAMB output (core flow as a function of time) is input to the SCAT code for calculation of blowdown heat transfer.

The LAMB results presented are:

- Core Average Inlet Flow Rate (normalized to unity at the beginning of the accident) following a Large Break.

##### 4.2 RESULTS OF THE SCAT ANALYSIS

This code completes the transient short-term thermal-hydraulic calculation for large breaks in jet pump reactors. The GEXL correlation is used to track the boiling transition in time and location. The post-critical heat flux heat transfer correlations are built into SCAT which calculates heat transfer coefficients for input to the core heatup code, CHASTE.

The SCAT results presented are:

- Minimum Critical Power Ratio Following a Large Break.
- Convective Heat Transfer Coefficient following a Large Break.

##### 4.3 RESULTS OF THE SAFE ANALYSIS

This code is used primarily to track the vessel inventory and to model ECCS performance during the LOCA. The application of SAFE is identical for all break sizes. The code is used during the entire course of the postulated accident, but after ECCS initiation, SAFE is used only to calculate reactor system pressure and ECCS flows, which are pressure dependent.



The SAFE results presented are:

- Water Level inside the Shroud (up to the time REFLOOD initiates) and Reactor Vessel Pressure

#### 4.4 RESULTS OF REFLOOD ANALYSIS

This code is used across the break spectrum to calculate the system inventories after ECCS actuation. The models used for the design basis accident (DBA) application ("DBA-REFLOOD") was described in a supplement to the SAFE code description transmitted to the USNRC December 20, 1974. The "non-DBA REFLOOD" analysis is nearly identical to the DBA version and employs the same major assumptions. The only differences stem from the fact that the core may be partially covered with coolant at the time of ECCS initiation and coolant levels change slowly for smaller breaks by comparison with the DBA. More precise modeling of coolant level behavior is thus requested principally to determine the contribution of vaporization in the fuel assemblies to the counter current flow limiting (CCFL) phenomenon at the upper tieplate. The differences from the DBA-REFLOOD analysis are:

- (1) The non-DBA version calculates core water level more precisely than the DBA version in which greater precision is not necessary.
- (2) The non-DBA version includes a heatup model similar to but less detailed than that in CHASTE, designed to calculate cladding temperature during the small break. This heatup model is used in calculating vaporization for the CCFL correlation, in calculating swollen level in the core, and in calculating the peak cladding temperature.

The REFLOOD results presented are:

- Water Level inside the Shroud
- Peak Cladding Temperature and Heat Transfer Coefficient for breaks calculated with small break methods

#### 4.5 RESULTS OF THE CHASTE ANALYSIS

This code is used, with suitable inputs from the other codes, to calculate the fuel cladding heatup rate, peak cladding temperature, peak local cladding oxidation, and core-wide metal-water reaction for large breaks. The detailed fuel model in CHASTE considers transient gap conductance, clad swelling and rupture, and metal-water reaction. The empirical core spray heat transfer and channel wetting correlations are built into CHASTE, which solves the transient heat transfer equations for the entire LOCA transient at a single axial plane in a single fuel assembly. Iterative applications of CHASTE determine the maximum permissible planar power where required to satisfy the requirements of 10CFR50.46 acceptance criteria.

The CHASTE results presented are:

- Peak Cladding Temperature versus time
- Peak Cladding Temperature versus Break Area
- Peak Cladding Temperature and Peak Local Oxidation versus Planar Average Exposure for the most limiting break size
- Maximum Average Planar Heat Generation Rate (MAPLHGR) versus Planar Average Exposure for the most limiting break size

A summary of the analytical results is given in Table 2. Table 3 lists the figures provided for this analysis. The MAPLHGR values for each fuel type for D2,3/QC1,2 are presented in Tables 4A through 4H.

#### 4.6 METHODS

In the following sections, it will be useful to refer to the methods used to analyze DBA, large breaks, and small breaks. For jet-pump reactors, these are defined as follows:

- a. DBA Methods. LAMB/SCAT/SAFE/DBA-REFLOOD/CHASTE. Break size: DBA.

- b. Large Break Methods (LBM). LAMB/SCAT/SAFE/non-DBA REFLOOD/CHASTE.  
Break sizes:  $1.0 \text{ ft}^2 \leq A < \text{DBA}$ .
- b. Small Break Methods (SBM). SAFE/non-DBA REFLOOD. Heat transfer coefficients: nucleate boiling prior to core uncover, 25 Btu/hr-ft<sup>2</sup>-°F after recovery, core spray when appropriate. Peak cladding temperature and peak local oxidation are calculated in non-DBA-REFLOOD. Break sizes:  $A \leq 1.0 \text{ ft}^2$ .

Table 2  
SUMMARY OF BREAK SPECTRUM RESULTS

<ul style="list-style-type: none"> <li>• Break Size</li> <li>• Location</li> <li>• <u>Single Failure</u></li> </ul>	<u>PCT (°F)</u>	<u>Peak Local Oxidation (%)</u>	<u>Core-Wide Metal-Water Reaction (%)</u>
<ul style="list-style-type: none"> <li>• 4.18 ft<sup>2</sup></li> <li>• Recirc Suction</li> <li>• LPCI Injection Valve</li> </ul>	2200 <sup>(1)</sup>	11.6	0.20

1. PCT from CHASTE

Table 3  
 LOCA ANALYSIS FIGURE SUMMARY - NON-LEAD PLANT

	<u>Large Break Methods</u>	
	Maximum Suction Break DBA (LPCI Injection Valve Failure) <u>(4.18 ft<sup>2</sup>)</u>	
Water Level Inside Shroud and Reactor Vessel Pressure	1	
Peak Cladding Temperature	2	
Heat Transfer Coefficient	3	
Core Average Inlet Flow	4	
Minimum Critical Power Ratio	5	-
Peak Cladding Temperature of the Highest Powered Plane Experiencing Boiling Transition	2	-
Variation with Break Area of Time for Which Hot Node Remains Uncovered	6	-

Table 4A  
 MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Dresden - 3

FUEL TYPE: 7D212 - NO GAD

<u>Average Planar Exposure (Mwd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
5,000	14.8	2192	0.036
12,500	14.7	2198	0.061
22,500	14.2	2189	0.082
30,000	12.5	1994	0.038

Table 4B  
 MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Dresden 2/Quad Cities 1,2

FUEL TYPE: 7D212L

<u>Average Planar Exposure (Mwd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
200	14.0	2198	0.042
1,000	14.1	2195	0.040
5,000	14.7	2198	0.037
10,000	14.7	2197	0.036
15,000	14.4	2198	0.081
25,000	13.2	2063	0.023
30,000	12.1	1938	0.015

Table 4C  
MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Dresden 2,3/Quad Cities 1

FUEL TYPE: 7D230

<u>Average Planar Exposure (MWd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
200	14.3	2198	0.034
1,000	14.5	2197	0.034
5,000	14.7	2198	0.032
10,000	14.5	2198	0.032
15,000	14.0	2198	0.074
20,000	13.7	2197	0.073
25,000	13.6	2198	0.070

Table 4D  
MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Quad Cities 1

FUEL TYPE: EEIC - PU

<u>Average Planar Exposure (MWd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
200	14.1	2198	0.042
1,000	14.3	2194	0.040
5,000	14.6	2197	0.039
10,000	14.2	2198	0.083
15,000	13.5	2198	0.112
20,000	13.3	2195	0.114
25,000	13.2	2195	0.116

Table 4E  
MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Dresden 2,3/Quad Cities 1,2

FUEL TYPE: 8D250

<u>Average Planar Exposure (Mwd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
200	11.2	2106	0.024
1,000	11.3	2109	0.024
5,000	11.9	2169	0.029
10,000	12.1	2179	0.029
15,000	12.2	2198	0.031
20,000	12.0	2199	0.031
25,000	11.5	2148	0.027
30,000	10.6	2020	0.017

Table 4F  
MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Dresden 2,3/Quad Cities 1,2

FUEL TYPE: 8D262

<u>Average Planar Exposure (Mwd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
200	11.1	2104	0.024
1,000	11.3	2107	0.024
5,000	11.9	2166	0.029
10,000	12.1	2175	0.029
15,000	12.2	2199	0.031
20,000	12.0	2199	0.032
25,000	11.6	2157	0.028
30,000	10.7	2042	0.019



Table 4G  
MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Dresden 2,3/Quad Cities 1,2

FUEL TYPE: 8DRB265L

<u>Average Planar Exposure (Mwd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
200	11.6	2189	0.035
1,000	11.6	2188	0.034
5,000	11.8	2198	0.034
10,000	11.9	2196	0.033
15,000	11.9	2198	0.034
20,000	11.7	2195	0.034
25,000	11.3	2154	0.030
30,000	10.7	2075	0.023

Table 4H  
MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Quad Cities 1

FUEL TYPE: Barrier LTA

<u>Average Planar Exposure (Mwd/t)</u>	<u>MAPLHGR (kW/ft)</u>	<u>PCT (°F)</u>	<u>Oxidation Fraction</u>
200	11.6	2171	0.032
1,000	11.6	2178	0.033
5,000	12.0	2198	0.034
10,000	12.1	2195	0.033
15,000	12.0	2200	0.033
20,000	11.7	2187	0.032
25,000	11.0	2115	0.056
30,000	10.5	2029	0.042

### 5. DESCRIPTION OF MODEL AND INPUT CHANGES

The only change between this ECCS analysis and the previous D2,3/QC1,2 analysis (Reference 12) is the use of the partial drill model. A description of this model is presented in Reference 13. Approval for the use of this model is given in Reference 14.

The addition of an alternate bypass flow path via holes drilled in the lower tieplate to the BWR 3's provides them with the same bypass flowpaths as the BWR4's (i.e., same as BWR4's with core plate holes plugged and holes drilled in the lower tie plates). Since the BWR3's do not currently include the LPCI modification, the ECC systems and the core configuration are the same as the BWR4 non-mod plants. The primary difference between these two groups of plants is that BWR3's have a lower power density than BWR4's. Thus, in this analysis credit is taken for flow through holes in the fuel assembly lower tieplates as in the lead plant (Duane Arnold).

6. CONCLUSIONS

The LOCA analysis in accordance with the requirements of Reference 2, for non-lead plants with loop selection logic and fuel bundles with drilled lower tieplates in a full or partial core loading, is presented in Figures 1 through 5. This analysis is for the maximum recirculation line suction break which is the most limiting break for this plant.

The characteristics that determine which is the most limiting break are:

- (a) the calculated hot node reflooding time,
- (b) the calculated hot node uncover time, and
- (c) the time of calculated boiling transition.

The time of calculated boiling transition increases with decreasing break size, since jet pump suction uncover (which leads to boiling transition) is determined primarily by the break size for a particular plant. The calculated hot node uncover time also generally increases with decreasing break size, as it is primarily determined by the inventory loss during the blowdown. The hot node reflooding time is determined by a number of interacting phenomena such as depressurization rate, counter current flow limiting and a combination of available ECCS.

The period between hot node uncover and reflooding is the period when the hot node has the lowest heat transfer. Hence, the break that results in the longest period during which the hot node remains uncovered results in the highest calculated PCT. If two breaks have similar times during which the hot node remains uncovered, then the larger of the two breaks will be limiting as it would have an earlier boiling transition time (i.e., the larger break would have a more severe LAMB/SCAT blowdown heat transfer analysis).

Figure 6 shows the variation with break size of the calculated time the hot node remains uncovered for D2,3/QC1,2. Based on these calculations, the DBA was

determined to be the break that results in the highest calculated PCT in the 1.0 ft<sup>2</sup> to DBA region. Although the 34% DBA break has a slightly longer total core uncovered time, the resulting PCT is less than the DBA. This is due to a much later boiling transition time associated with the 34% DBA and a later core uncover time which results in a slower heatup rate due to a lower decay heat.

For breaks smaller than 1.0 ft<sup>2</sup> the calculated PCT's will be less than those calculated in the 1.0 ft<sup>2</sup> to DBA break range, as has been demonstrated for the lead plant (Reference 11) for D2,3/QC1,2.

The single failure evaluation showing the remaining ECCS following an assumed failure and the effects of a single failure or operator error that causes any manually controlled, electrically operated valve in the ECCS to move to a position that could adversely affect the ECCS are presented in Reference 15.

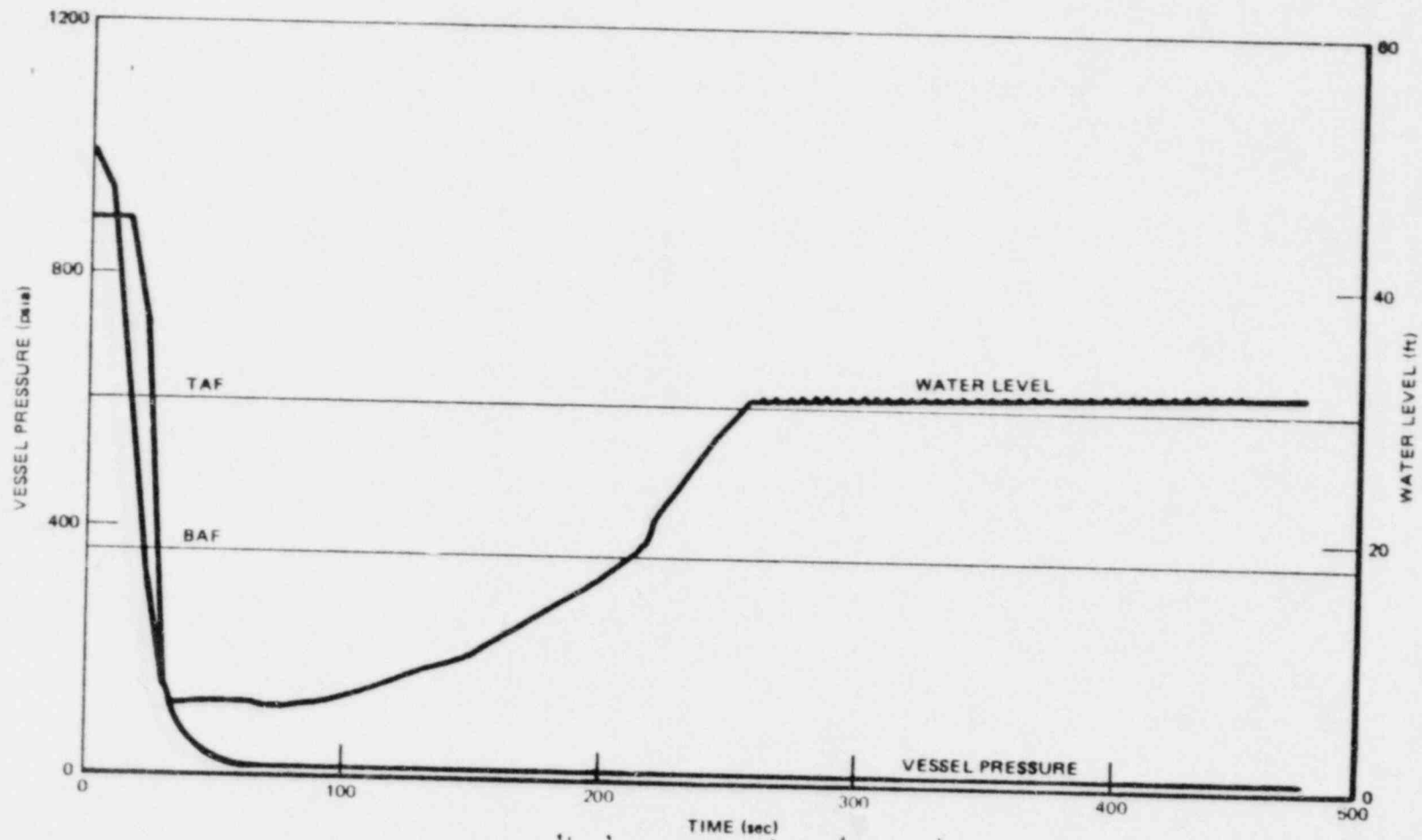


Figure 1. Water Level Inside the Shroud and Reactor Vessel Pressure Following a Recirculation Suction Line Break, LPCI Injection Valve Failure, Break Area = 4.18 ft<sup>2</sup> (LBM)

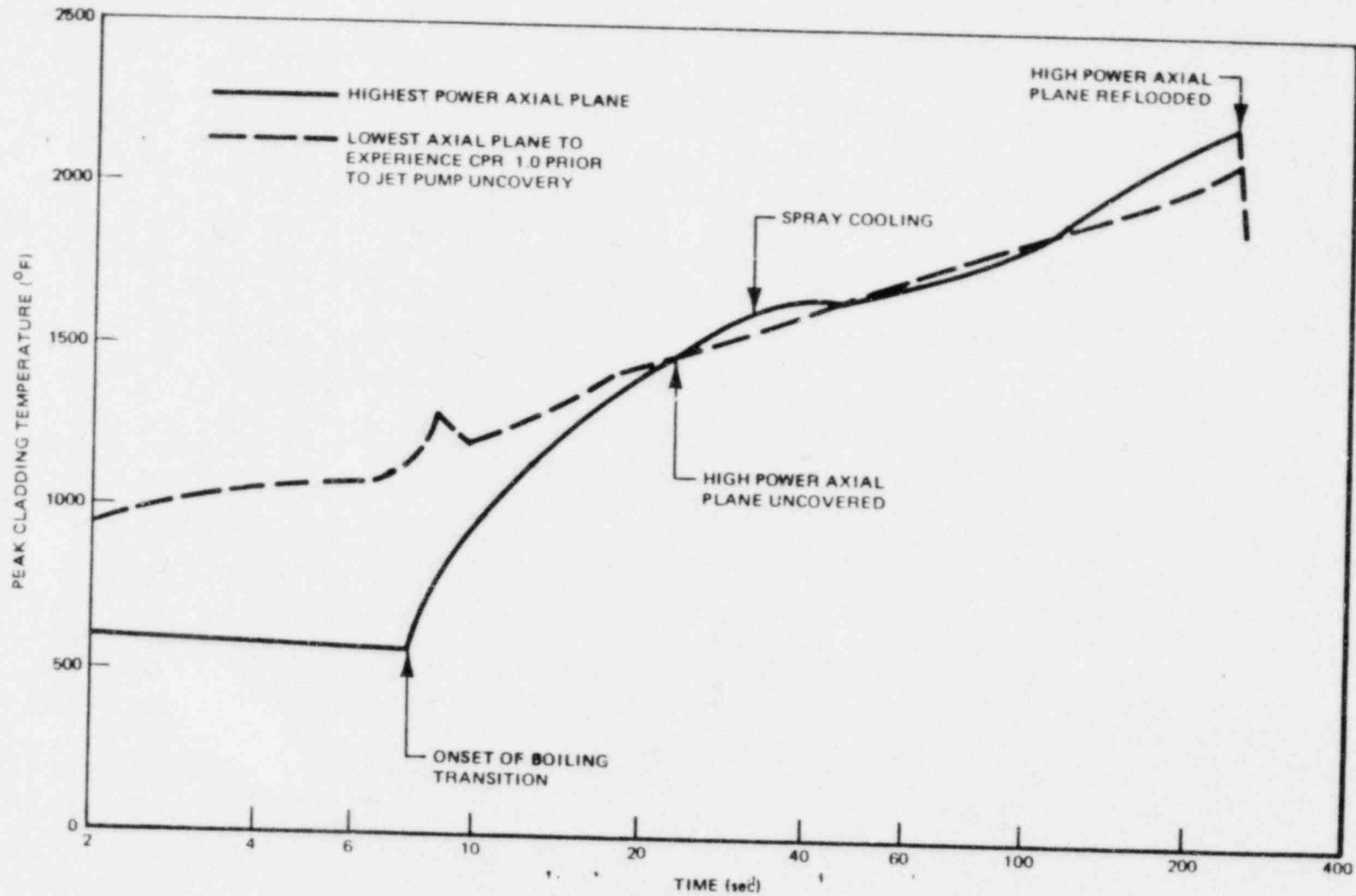


Figure 2. Peak Cladding Temperature Following a 4.18 ft<sup>2</sup> Recirculation Line Break, LPCI Injection Valve Failure (DBA)

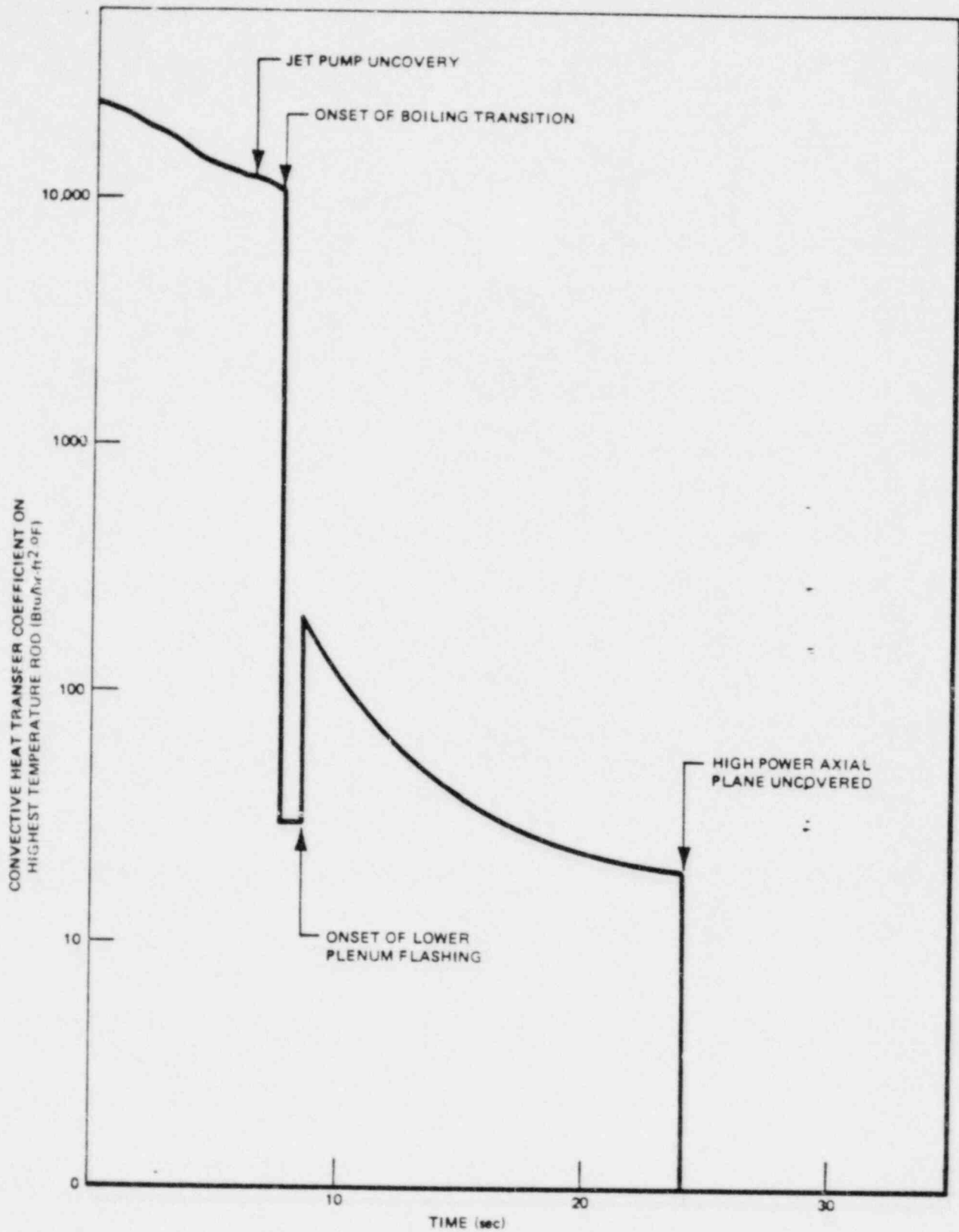


Figure 3. Fuel Rod Convective Heat Transfer Coefficient During Blowdown at the High Power Axial Node for a  $4.18 \text{ ft}^2$  Recirculation Line Suction Break (DBA)

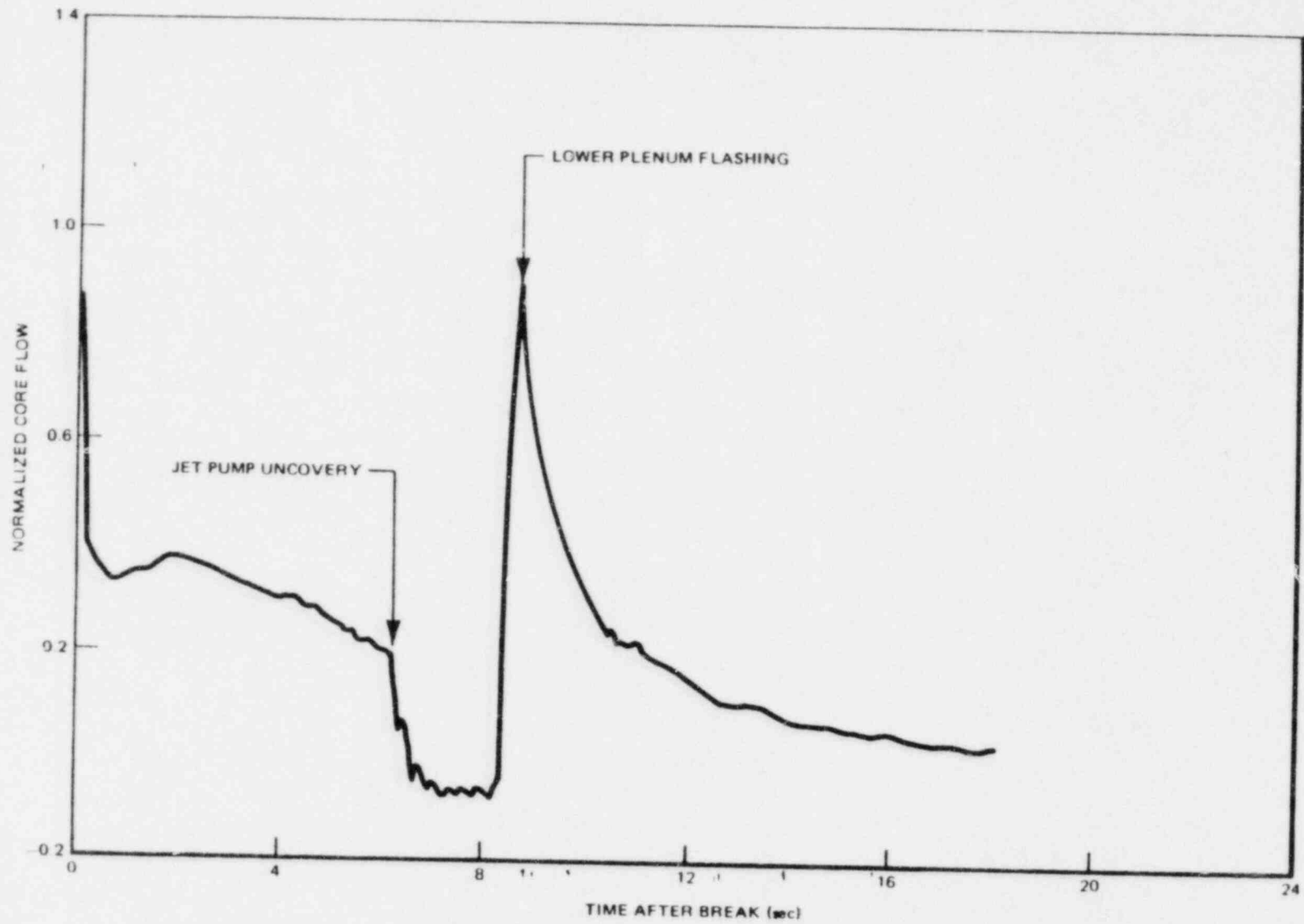
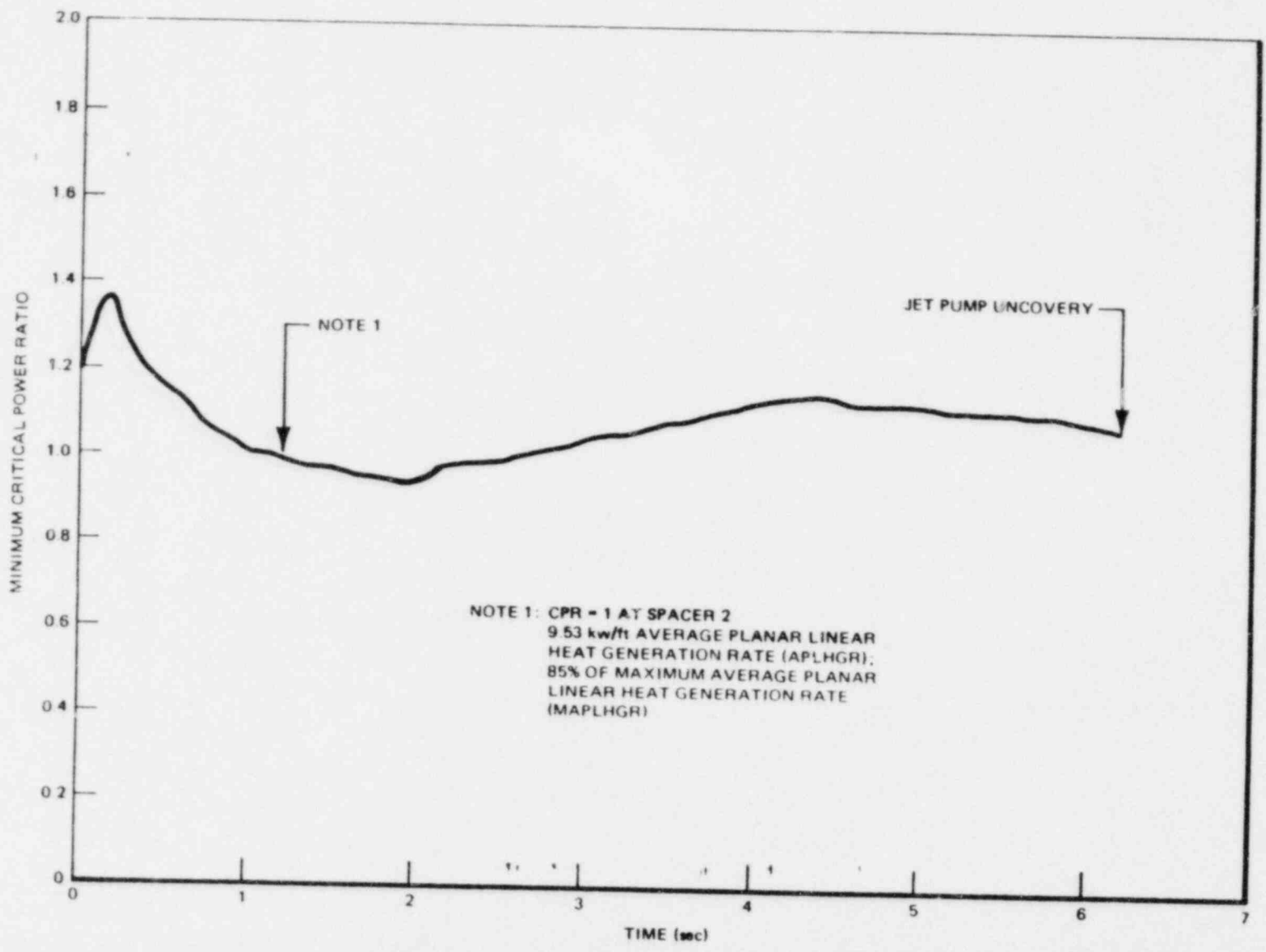


Figure 4. Normalized Core Average Inlet Flow Following a 4.18 ft<sup>2</sup> Recirculation Line Suction Break (DBA)



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Figure 5. Minimum Critical Power Ratio Following a 4.18 ft<sup>2</sup> Recirculation Line Suction Break (DBA)