DEC 27 1973

Docket Nos. 50-237 and 50-249

Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Nuclear Licensing Administrator -
Boiling Water Reactors
Post Office Box 767
Chicago, Illinois 60690

Change No. 25
License No. DPR-19
Change No. 15
License No. DPR-25

Gentlemen:

Your letters of September 14, 1973, and October 17, 1973, proposed changes to the Technical Specifications of Dresden Units 3 and 2, respectively, to revise the maximum reactivity that could be added by the dropout of any insequence control blade and to add surveillance requirements for the rod worth minimizer. Background information and analyses for the proposed Technical Specification revisions were submitted by your letters dated August 3, 1972, May 2, 1973, and August 13, 1973. Your letters of August 1, 1973, and October 17, 1973, proposed changes to the Technical Specifications of Dresden Units 3 and 2, respectively, to require faster control rod scram times. Analyses and additional information for the proposed control rod scram insertion times were provided with your letters dated July 2, 1973, and October 18, 1973. We have reviewed all of these submittals and are combining all of the proposed changes into this single authorization.

During our review of your proposed changes dated September 14 and October 17, 1973, regarding control rod reactivity worth, we informed your staff that certain modifications were necessary in the bases for the maximum reactivity addition specification and the limiting condition for operation of the rod worth minimizer. The modification to the bases has been made. The modification to the limiting condition of operation for rod worth minimizer operability has been made with an effective date of June 1, 1974, to allow time for implementation of measures necessary to achieve acceptable rod worth minimizer reliability and operability. With these modifications, we conclude that the revised specifications limit the consequences of an insequence rod drop, and the probability of an out-of-sequence rod drop to an acceptably low level.

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Your letters of August 1, 1973, and October 17, 1973, requested a change to the Technical Specifications for Dresden Units 3 and 2, respectively, as the result of your reanalysis of the reactivity worth of control rods as a function of their position in the reactor core. Your analysis shows that the rate of loss of reactivity during the early part of control rod motion during scram is not as great as previous analyses have indicated. The change in the rate of scram has its greatest effect on two transients previously considered, turbine trip without available steam bypass and main steam isolation valve closure with high neutron flux scram. These two transient analyses are used to evaluate the adequacy of relief valves and safety valves to limit primary system pressure transients. The analyses show that with two provisions, the relief and safety valves can maintain design pressure margins even with the changed scram reactivity response. These provisions are: (1) a requirement for more rapid control rod scram insertion times, and (2) a reduction in reactor power beyond a certain burnup in the operating cycle. For Dresden 2, you estimate that beyond 6550 MWd/ton core average exposure, the power would have to be limited to 97% of full power to assure that there will be no change in the margin of safety following the turbine trip without bypass transient. Our understanding is that you have derated to 97% power at 6200 MWd/ton as you proposed in your letter. For Dresden 3, derating to 97% of power occurred at about 750 MWd/ton exposure increment in cycle 2. You estimate that no further derating of Dresden 3 is likely because of a planned refueling outage in February 1974. It is our understanding that Dresden 2 might have to further derate tc 91% power at about 9200 MWd/ton into the present cycle which would occur about August 1974. Your confirmation of our understanding of the Dresden 2 derating is requested. If you intend to exceed a burnup of 9200 MWd/ton during the current fuel cycle at Dresden 2, you are requested to provide us with a revised analysis scram reactivity at least 90 days prior to reaching the 9200 MWd/ton incremental burnup.

We have concluded, subject to the above power restrictions, that the proposed changes to the Technical Specifications, as modified, do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered. A copy of our Safety Evaluation regarding rod drop accident considerations is enclosed.

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Pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating Licenses Nos. DPR-19 and DPR-25 are hereby changed by replacing pages 13, 20, 57, 58, 62, 63 and 64 with the revised pages 13, 20, 57, 57a, 58, 62, 62a, 62b, 63 and 64 appended hereto.

Sincerely,

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Revised pages

cc w/enclosures:
John W. Rowe, Esquire
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza
Chicago, Illinois 60670

Morris Public Library
604 Liberty Street
Morris, Illinois 60670

cc w/enclosures and CEC's ltrs dtd
8/3/72, 5/2/73, 7/2/73, 8/13/73,
9/14/73, 10/17/73 and 10/18/73:
Mr. Hans L. Hamester (2 cys)
ATTN: Joan Sause
Office of Radiation Programs
Environmental Protection Agency
Room 647A East Towers, Waterside Mall
401 M Street, S. W.
Washington, D. C. 20460

Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
1 N. Wacker Drive
Chicago, Illinois 60606

Distribition
Docket File
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PCollins
SKari
SVarga

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OFFICE	L:ORB #2	L:ORB #2	L:ORB #2	L:ORB #2	L:ORB	
SURNAME	RDSilver:rwg	RMDiggs	RWReid	DLZiemann	DJSkovholt	
DATE	12/26/73	12/26/73	12/26/73	12/27/73	12/27/73	

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

COMMONWEALTH EDISON COMPANY

DOCKET NOS. 50-237 AND 50-249

ROD DROP ACCIDENT

In letters dated September 14, 1973, and October 17, 1973, Commonwealth Edison Company submitted requests for changes to the Technical Specifications for the Dresden 3 and 2 reactors, respectively, concerning the rod drop accident. These changes are based on new calculational models developed by the General Electric Company, presented in references 1, 2, and 3, and on a change in the assessment of the accident and scram reactivity shape. These changes result in a reduction in maximum allowable insequence control rod reactivity worth from 2.5% to 1.3% delta k/k, and increase the assurance that a control rod is not in an out-of-sequence position during low power operation.

The rod drop accident is one of the design basis accidents for boiling water reactors. For calculational purposes, it is assumed that a control rod blade separates from its drive, lodges in the core with the drive withdrawn and drops at the time which causes the most serious power excursion due to rapid reactivity insertion. The consequences of this accident are evaluated by determining the energy input to the fuel assuming that the reactivity worth of the dropped rod is the maximum which could occur. The maximum acceptable energy in the fuel is limited such that, in the event of fuel cladding failure, the energy input into the coolant will not result in a pressure pulse which might damage the core geometry or the reactor pressure vessel.

The analytical methods used by the General Electric Company (GE) to evaluate the consequences of the rod drop accident have been reviewed by the staff and independent calculations have been performed by Brookhaven National Laboratory which show reasonable agreement with GE results. Based on these reviews, it is concluded that the analytical methods used by GE are acceptable.

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(1) Paone, C. J., Stirn, R. C., and Wooley, J. A., "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527, March 1972.
(2) Stirn, R. C., Paone, C. J., and Young, R. M., "Rod Drop Accident Analysis for Large BWR's", Supplement 1 - NEDO-10527, July 1972.
(3) Stirn, R. C., Paone, C. J., and Haun, J. M., "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores", Supplement 2 - NEDO-10527, January 1973.

Application of the GE analytical methods to operating reactors requires that the input parameters conservatively represent the reactor core over a broad range of operating conditions. The proposed changes to the Technical Specifications include, in the Bases, a set of boundary conditions which are used to calculate the maximum allowable reactivity worth of control rod. It is not expected that these boundary conditions will be exceeded for reactor cores of current design. The boundary conditions include a maximum inter-assembly local power peaking factor, an end-of-cycle delayed neutron fraction, a beginning of life Doppler reactivity feedback, the technical specification control rod scram insertion rate, a control rod drop velocity of 3.11 ft/sec, and specified accident and scram reactivity shape functions. The rod drop velocity of 3.11 ft/sec is based on tests with a "worse case" rod built with maximum clearances and features known to contribute to the high rod drop velocities. The difference between the mean rod drop velocity and the 99.9% confidence limit for a group of production rods was added to the mean velocity obtained for the "worst case" control rod. We have added the value of 0.005 for the end-of-cycle delayed neutron fraction to further define the boundary assumptions. In addition, we have added a statement to the bases that each reload core must be analyzed to show conformance to the bounding assumption. The peak fuel enthalpy resulting from an insequence rod drop accident within the above boundary conditions is calculated not to exceed 280 cal/gram, which is acceptably below the peak fuel enthalpy at which prompt fuel dispersal would occur based on the SPERT tests. Based on the above, the resultant maximum allowable insequence rod worth of 1.3% delta k/k is acceptable.

Separate consideration is being given to the potentially adverse effect on the rod drop accident compaction of boron carbide in the control rods in the event of inverted poison tubes. The evaluation of the effect of possible inverted poison tubes on the allowable insequence rod worth is currently in progress and if determined necessary, appropriate changes to the allowable control rod reactivity worth will be made.

If a control rod is withdrawn out-of-sequence, a rod worth of greater than 1.3% delta k/k could result. In the event of rod drop accident associated with such an out-of-sequence rod, the peak fuel enthalpy could exceed 280 cal/gram. The rod worth minimizer (RWM) is designed as an operator aid to prevent an out-of-sequence rod withdrawal. Current Technical Specifications allow the RWM to be bypassed if it is inoperable during a reactor startup provided that a second operator is assigned to monitor the rod withdrawal sequence. To increase the control on RWM availability during reactor startups, the technical specification is being changed to require that the RWM be operable for the withdrawal of a significant number of control rods. The effective date of this change in technical specifications is being deferred concerning RWM operability until June 1, 1974, to allow any necessary upgrading of the RWM to be accomplished.

Based on the above, we conclude that the proposed changes do not present a significant hazards consideration, and there is reasonable assurance that the health and safety of the public will not be endangered.

Richard D. Silver
Operating Reactors Branch #2
Directorate of Licensing

Robert W. Reid
Operating Reactors Branch #2
Directorate of Licensing

J. Kneeland for
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: December 27, 1973

Bases:

2.1 The transients expected during operation of the Dresden 2 and 3 units have been analyzed starting at the rated thermal power condition of 2527 MWt at 100% recirculation flow. It should be noted that this power is equivalent to the designed maximum power and a higher power cannot physically be obtained under normal operating conditions unless the turbine bypass system is used. In addition, 2527 MWt is the licensed maximum steady state power level of Dresden Units 2 and 3. This maximum steady state power will never be knowingly exceeded.

Dresden Units 2 and 3 were not analyzed from a power level which included instrument errors. To protect against misleading conclusions from analysis not reflecting realistic instrument errors, conservatism was incorporated by conservatively estimating the controlling factors such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, axial power shapes, etc. These factors are all selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model evolved over many years, has been substantiated in operation as a conservative tool for the evaluation of reactor dynamics performance. Comparisons have been made showing results obtained from a General Electric boiling water reactor and the predictions made by the model. The comparisons and results are summarized in Topical Report APED-5698, "Summary of Results Obtained from A Typical Startup and Power Test Program for a General Electric Boiling Water Reactor."

The void reactivity coefficient utilized in the analysis is conservatively estimated to be about 25% larger than the most negative value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to the scram worth of about 75% of the control rods. The scram delay time and rate of rod insertion are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The insertion of the first dollar of reactivity strongly turns the transient and the stated 5% and 20% insertion time conservatively accomplishes this desired initial effect. The time for 50% and 90% insertion are given to assure proper completion of the insertion stroke, to further assure the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady state condition.

The design peaking factors at the full power conditions for Dresden 2 and 3 result in a MCHFR value of 2.04. For analysis of the thermal consequences of the transients, higher peaking factors are used, such that a MCHFR of 1.9 is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the rated power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels. As an example, consider the sensitivity analyses conduct to provide the answer to Question 4.6.4 of Amendment 7 of the Dresden Unit 2 SAR. From the results of the Case 1 transient, the turbine trip with flux scram without bypass or relief, a significant reduction in the neutron flux and heat flux peaks will be realized when the smaller void reactivity coefficient is used. For this particular transient, if it were also analyzed at a power level of 110% of rated but with the expected void reactivity coefficient, the resulting heat flux peak would be less than the peak resulting from the analysis

Bases:

1.2 The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, coolant system piping and isolation condenser. The respective design pressures are 1250 psig at 575°F, 1175 psig at 560°F, and 1250 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and isolation condenser and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the primary coolant system.

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationship of stress levels to yield strength are comparable for the isolation condenser and primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1000 psig. For the turbine trip or loss of electrical load transients the turbine trip scram or generator load rejection scram, together with the turbine bypass system limit the pressure to approximately 1100 psig (4). In addition, pressure relief valves have been provided to reduce the probability of the safety valves operating in the event that the turbine bypass should fail. These valves and the neutron flux scram limit the reactor pressure to 1185 psig (5) and (6) which is 25 psi below the setting of the first safety valve. Finally, the safety valves are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for the relief valves or turbine bypass system. Credit is taken for the neutron flux scram however.

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full scale pressure recorder.

(4) SAR Section 11.2.2.
(5) SAR Section 4.4.3.
(6) Special Report No. 29.

3.3 LIMITING CONDITION FOR OPERATION

3. (a) Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would not make the core more than 0.013 delta K supercritical.

(b) Whenever the reactor is in the startup or run mode below 10% rated thermal power, no control rods shall be moved unless the rod worth minimizer is operable or a second independent operator or engineer verifies that the operator at the reactor console is following the control rod program. After June 1, 1974, the second operator may be used as a substitute for an inoperable rod worth minimizer during a startup only if the rod worth minimizer fails after withdrawal of at least twelve control rods.

4.3 SURVEILLANCE REQUIREMENTS

3. (a) To consider the rod worth minimizer operable, the following steps must be performed:

(i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct.

(ii) The rod worth minimizer computer on-line diagnositc test shall be successfully completed.

(iii) Proper annunciation of the select error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

(iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.

(b) If the rod worth minizer is inoperable while the reactor is in the startup or run mode below 10% rated thermal power second independent operator or engineer is being used, he shall verify that all rod positions are correct prior to commencing withdrawal of each rod group.

3.3 LIMITING CONDITIONS FOR OPERATION

4. Control rod shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operating with limiting control rod patterns, as determined by the nuclear engineer, either:

   a. Both RBM channels shall be operable; or

   b. Control rod withdrawal shall be blocked; or

   c. The operating power level shall be limited so that the MCHFR will remain above 1.0 assuming a single error that results in complete withdrawal of any single operable control rod.

4.3 SURVEILLANCE REQUIREMENTS

4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have been observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

3.3 LIMITING CONDITION FOR OPERATION	4.3 SURVEILLANCE REQUIREMENT

3.3 LIMITING CONDITION FOR OPERATION

C. Scram Insertion Times

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

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	5.00

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:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	5.300

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

4.3 SURVEILLANCE REQUIREMENT

C. Scram Insertion Times

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

2. At 16 week intervals, 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% of the control rod drives have been scram tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. 25 of the operable control rods, selected to be uniformly distributed throughout the core, shall be scram-time tested at full reactor pressure at the time intervals listed below following any outage exceeding 72 hours in duration: 1 week, 2 weeks, 4 weeks, 8 weeks, 16 weeks and continuing at 16 week intervals:
   a) If the mean 90% insertion time of the tested control rod drives increases by more than 0.25 seconds or if the mean insertion time exceeds 3.5 seconds, then an additional sample of 25 control rods, selected to be uniformly distributed throughout the core, shall be scram tested. If the mean 90% insertion time of the 50 selected control rod drives exceeds 4.25 seconds, then all operable drives will be tested. Subsequent testing shall revert to the original 25 control rods at the 1 week, 2 week, etc., sequence interval; and

indicative of a generic control rod drive problem and the reactor will be shutdown.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in the SAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would indicate an uncoupled condition.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 delta K supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident.(3) These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM. This 0.013 delta K limit, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 1.

The analysis of the control rod drop accident was orginally presented in Sections 7.9.3, 14.2.1.2 and 14.2.1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

Bases (cont'd)

These techniques are described in a topical report [1] and two supplements. [2] [3]

By using the analytical models described in those reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 10% of rated power, the specified limit on insequence control rod or control rod segment worths will limit the peak fuel enthalpy to less than 280 cal/gm. Above 10% power even single operator errors cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalpy of 280 cal/gm should a postulated control rod drop accident occur.

(1) Paone, C.J., Stirn, R.C. and Wooley, J.A., "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527, March 1972.

(2) Stirn, R.C., Paone, C.J., and Young, R.M., "Rod Drop Accident Analysis for Large BWR's", Supplement 1 - NEDO-10527, July 1972

(3) Stirn, R.C., Paone, C.J., and Haun, J.M., "Rod Drop Accident Analysis for Large BWR's Addendum No. 2, Exposed Cores", Supplement 2-NEDO 10527, January 1973.

The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specified 0.013 K limit on insequence control rod or control rod segment worths. Details of this analysis are contained in Reference 4. Each core reload will be analyzed to show conformance to the limiting parameters.

a. A maximum inter-assembly local power peaking factor of 1.30 or less. [5]

b. An end-of-cycle delayed neutron fraction of 0.005.

c. A beginning-of-life Doooper reactivity feedback.

d. The technical specification rod scram insertion rate.

e. The maximum possible rod drop velocity (3.11 ft./sec.)

f. The design accident and scram reactivity shape function.

g. The minimum moderator temperature to reach criticality.

(4) Exhibit A attached to September 14, 1973 letter from Byron Lee, Commonwealth Edison Company, to J. F. O'Leary, U.S. Atomic Energy Commission.

(5) To include the power spike effect caused by gaps between fuel pellets.

It is recognized that these bounds are conservative with respect to expected operating conditions. If any one of the above conditions is not satisfied, a more detailed calculation will be done to show compliance with the 280 cal/gm design limit.

In most cases the worth in insequence rods or rod segments will be substantially less than 0.013ΔK. Further, the addition of 0.013ΔK worth of reactivity as a result of a rod drop and in a 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 the 280 cal/gm design limit. However, the 0.013ΔK 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/gm less 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 10CFR 100. For 8 x 8 fuel, less 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. Ref. Section 7.9 SAR. 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 functions of the Rod Worth Minimizer. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 10% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is requried to be operating during a startup for the withdrawal of a significant number control rods for any startup after June 1, 1974.

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 are 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 level 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 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. Amendments 17/18 and 19/20 present the results of an evaluation of a rod block monitor failure. These amendments show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCHFR's less than 1.0. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability 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 then limiting patterns.

C. **Scram Insertion Times**

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCHFR 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 MCHFR remains greater than 1.0. Figure 3.5.2 of the SAR (1) shows the control rod scram reactivity used in analyzing the transients. Figure 3.5.2 (1) should not be confused with the total control rod worth, 18% k, as listed in some amendments to the SAR. The 18% k value represents the amount of reactivity available for withdrawal in the cold clean core, whereas the control rod worths shown in Figure 3.5.2 of the SAR (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 absberved time delay of about 270 milliseconds.

(1) For Cycle 2 of Dresden 3 and Cycle 3 of Dresden 2 Figure I-1 of Special Report No. 29

Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid de-energizes. Approximately 200 milliseconds later, control rod motion begins. The time to de-energize 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. Plugging of the internal drive filters has resulted in occasional increases in scram times at rates greater than one second per week of startup operation. Scram times of new drives are approximately 2.5 to 3.0 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 3.3.0. The program was developed on the basis of the statistical approach outlined below and judgement.

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 exceed 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 3.3.C greatly exceeds the expected range, this sampling technique gives assurance that the limits of 3.3.C will not be exceeded. As further assurance that the limits of 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 exceed 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

(1) For Cycle 2 of Dresden 3 and Cycle 3 of Dresden 2, Figure I-1 of Special Report No. 29