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December 6, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Generating Station Units 1 and 2
Braidwood Generating Station Units 1 and 2
Dropped Rod Reanalysis
NRC Docket Nos. 50-454, 50-455, 50-456, and 50-457

References (a): December 29, 1979, letter from D. L. Peoples to J. G. Keppler.

(b): May 13, 1983 memorandum from D. G. Eisenhut to NRC Commissioners (Board Notification No. 83-66).

(c): September 29, 1983 Memorandum from D. G. Eisenhut to NRC Commissioners (Board Notification No. 83-66A).

(d): April 3, 1980, Memorandum from S. A. Varga to ASLB's (BN-79-18).

Dear Mr. Denton:

This letter provides advance copies of revised Byron/Braidwood FSAR pages which document the reanalysis of a dropped full-length control rod. This reanalysis demonstrates that the interim control rod withdrawal limits contemplated in Section 15.2.4.3 of the Byron SER will not be required.

Westinghouse advised the NRC in November, 1979 of a deficiency in the rod drop analyses which had been performed for certain plants. Commonwealth Edison notified the NRC in reference (a) that this issue involved the Byron and Braidwood units. The Byron ASLB was initially informed of this issue in reference (d). According to references (b) and (c) the NRC has now completed the review of a Westinghouse topical report (WCAP-10297(P)) on the revised dropped rod methodology for flux rate trip plants.

Attachment A to this letter contains revised Byron/Braidwood FSAR pages which incorporate the results of a reanalysis of rod drop transients using the revised methodology. These pages will be incorporated into the FSAR at the earliest opportunity.

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This revised analysis provides the basis for operating the reactors without rod withdrawal limitations. Since it is part of the licensing basis, the continuing validity of the input assumptions will be verified as part of future 50.59 evaluations such as those performed for each reload.

Please address questions regarding this matter to this office.

One signed original and fifteen copies of this letter and the attachments are provided for NRC use.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

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ATTACHMENT A

Revised FSAR Pages Incorporating New

Dropped Rod Analysis

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15.4.2.3 Radiological Consequences

There are only minimal radiological consequences associated with an uncontrolled rod cluster control assembly bank withdrawal at power event. The reactor trip causes a turbine trip, and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage is postulated to occur, the radiological consequences associated with atmospheric steam release from this event are less severe than the steamline break accident analyzed in Subsection 15.1.5.

15.4.2.4 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the limit value. The radiological consequences would be less severe than the steamline break accident analyzed in Subsection 15.1.5.

15.4.3 ROD CLUSTER CONTROL ASSEMBLY MISOPERATION (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misoperation accidents include:

a. One or more dropped RCCA's within the same group,

b. A dropped RCCA bank,

c. Statically misaligned RCCA,

d. Withdrawal of a single RCCA.

Each RCCA has a position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

Full length RCCA's are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCA's is divided into two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second

group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCA's of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

The dropped assemblies, dropped assembly bank, and statically misaligned assembly events are classified as ANS Condition II incidents (incidents of moderate frequency) as defined in Section 15.0.1. The single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single rod cluster control assembly (RCCA) from the inserted bank at full power operation. The operator could deliberately withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event analyzed must result from multiple wiring failures (probability for single random failure is on the order of 10^{-4}/year-refer to Section 7.7.2.2) or multiple deliberate operator actions and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered so low that the limiting consequences may include slight fuel damage.

Thus, consistent with the philosophy and format of ANSI N 18.2, the event is classified as a Condition III event. By definition "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant", and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged..."

This selection of criterion is not in violation of GDC 25 which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any <u>single</u> malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control <u>rods</u>." (Emphases have been added). It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that the criterion established for the single rod withdrawal at power is appropriate and in accordance with GDC 25.

A dropped assembly or assembly bank is detected by:

a. Sudden drop in the core power level as seen by the nuclear instrumentation system;

b. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

c. Rod at bottom signal;

d. Rod deviation alarm;

e. Rod position indication.

Misaligned assemblies are detected by:

a. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

b. Rod deviation alarm;

c. Rod position indicators.

The resolution of the rod position indicator channel is $\pm 5\%$ of span ($\pm$ 7.2 inches). Deviation of any assembly from its group by twice this distance (10 % of span, or 14.4 inches) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5% of span. If the rod deviation alarm is not operable, the operator is required to take action as required by the technical specifications.

If one or more rod position indicator channels should be out of service, detailed operating instructions shall be followed to assure the alignment of the non-indicated assemblies. The operator is also required to take action as required by the technical specifications.

In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core

power, and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature T reactor trip, although due to the increase in local power density it is not possible in all cases to provide assurance that the core safety limits will not be violated.

Plant systems and equipment which are available to mitigate the effects of the various control rod misoperations are discussed in Section 15.0.8 and listed in Table 15.0-7. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

15.4.3.2 Analysis of Effects and Consequences

a. Dropped RCCA's, dropped RCCA bank, and statically misaligned RCCA.

Method of Analysis

1. One or more dropped RCCA's from the same group.

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC code. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 9.

2. Dropped RCCA Bank

A LOFTRAN calculation is not necessary for the dropped RCCA event because reactor trip occurs within approximately 2.5 seconds.

3. Statically Misaligned RCCA.

Steady state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then used as input to the THINC code to calculate the DNBR.

Results

1. One or more dropped RCCA's.

   Single or multiple dropped RCCA's within the same group result in a negative reactivity insertion which may be detected by the power range negative neutron flux rate trip circuitry. If detected, the reactor is tripped within approximately 2.5 seconds following the drop of the RCCA's. The core is not adversely affected during this period, since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed. The operator may manually retrieve the RCCA by following approved operating procedures.

   For those dropped RCCA's which do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

   For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figure 15.4-12a shows a typical transient response to a dropped RCCA (or RCCA's) in automatic control. Uncertainties in the initial condition are included in the DNB evaluation as described in Reference 9. In all cases, the minimum DNBR remains above the limit value.

2. Dropped RCCA Bank.

   A dropped RCCA bank typically results in a reactivity insertion greater than 500 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is tripped within approximately 2.5 seconds following the drop of a RCCA bank. The core is not adversely affected during this period, since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed to further cool down the plant. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following the incident.

3. Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCAA is fully inserted, or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The insertion limits in the technical specifications may vary from time to time depending on a number of limiting criteria. It is preferable, therefore, to analyze the misaligned RCCA case at full power for a position of the control bank as deeply inserted as the criteria on minimum DNBR and power peaking factor will allow. The full power insertion limits on control bank D must then be chosen to be above that position and will usually be dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For this RCCA misalignment, with bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limit value. This case was analyzed assuming the initial reactor power, pressure, and RCS temperatures were at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in initial conditions were included as described in WCAP-8567.

DNB calculations have not been performed specifically for assemblies missing from other banks; however, power shape calculations have been done as required for the RCCA ejection analysis. Inspection of the power shapes shows that the DNB and peak kW/ft situation is less severe than the bank D case discussed above assuming insertion limits on the other banks equivalent to a bank D full-in insertion limit.

For the RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit

value. This case was analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in the initial conditions are included as described in WCAP-8567.

DNB does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of a RCCA group misalignment condition by the operator, the operator is required to take action as required by the plant technical specifications and operating instructions.

b. Single RCCA Withdrawal

Method of Analysis

Power distributions within the core are calculated using the computer codes as described in Table 4.1-2. The peaking factors are then used by THINC to calculate the minimum DNBR for the event. The case of the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power, was analyzed. This incident is assumed to occur at beginning-of-life since this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

Results

For the single rod withdrawal event, two cases have been considered as follows:

1. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature,

and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in Subsection 15.4.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBR's than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNBR from falling below the limit value. Evaluation of this case at the power and coolant conditions at which the overtemperature ΔT trip would be expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the limit value is 5%.

2. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA will result in the immobility of the other RCCA's in the controlling bank. The transient will then proceed in the same manner as Case 1 described above.

For such cases as above, a reactor trip will ultimately ensue, although not sufficiently fast in all cases to prevent a minimum DNBR in the core of less than the limit value. Following reactor trip, normal shutdown procedures are followed.

15.4.3.3 Radiological Consequences

The most limiting rod cluster control assembly misoperation, accidental withdrawal of a single RCCA, is predicted to result in limited fuel damage. The subsequent reactor and turbine trip would result in atmospheric steam dump, assuming the condenser was not available for use. The radiological consequences from this event would be no greater than the main steamline break event, analyzed in Subsection 15.1.5.

15.4.3.4 Conclusions

For cases of dropped RCCA's or dropped banks, for which the reactor is tripped by the power range negative neutron flux rate trip, there is no reduction in the margin to core thermal limits, and consequently the DNB design basis is met. It is shown for all cases which do not result in reactor trip that the DNBR remains greater than the limit value and, therefore, the DNB design is met.

For all cases of any RCCA fully inserted, or bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the limit value.

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

15.4.4 STARTUP OF AN INACTIVE REACTOR COOLANT PUMP AT AN INCORRECT TEMPERATURE

15.4.4.1 Identification of Causes and Accident Description

If the plant is operated with one or two loop stop valves of one of its loops closed, there is no flow from the reactor vessel and active loops to inactive loop and the plant operates much as if it were a three-loop unit. In such a situation the isolated section of the loop could be cooler than the temperature of the active loops.

Interlocks are provided to ensure that an accidental startup of an isolated loop which has a lower temperature than the core and active loops will be a relatively slow accident. Boron concentration in the isolated loop is administratively controlled. The interlocks ensure that flow from the isolated loop to the remainder of the RCS takes place through the relief line bypassing the cold-leg stop valve for a period of approximately 3 hours before the cold-leg stop valve can be opened. The flow through the relief line is made low (no more than 182 gpm) so that the temperature and boron concentration in the isolated loop are brought to equilibrium with the remainder of the system at a relatively slow rate should the administrative procedures be violated and an attempt made to open stop valves when the isolated loop temperatures or boron concentration is lower than that in the core and active loops. These conditions apply for Modes 1, 2, 3 and 4. For Modes 5 and 6, the temperature and boron concentration considerations apply, but the 3 hour minimum flow condition does not.

Interlocks are provided to:

a. Prevent opening of a hot leg loop stop valve unless the cold leg stop valve in the same loop is fully closed.

b. Prevent starting a reactor coolant pump unless:

   1. The cold leg loop stop valve in the same loop is fully closed and the loop bypass valve is fully open, or

15.4.9 References

1. Risher, D. H., Jr. and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kenetics Computer Code," WCAP-7979-A (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975.

2. Hargrove, H. G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972.

3. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, June 1972.

4. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

5. Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

6. Taxelius, T. G. (Ed), "Annual Report - SPERT Project, October, 1968, September, 1969," Idaho Nuclear Corporation IN-1370, June 1970.

7. Liimataninen, R. C. and Testa, F. J., "Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements," ANL-7225, January - June 1966, p. 177, November 1966.

8. Bishop, A. A., Sanburg, R. O. and Tong, L. S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

9. Morita, T., Osborne, M. P., Loftus, P. A., and Woodcock, J., "Dropped Rod Methodology for Negative Flux Rate Trip Plants," WCAP 10297-P-A, June 1983.

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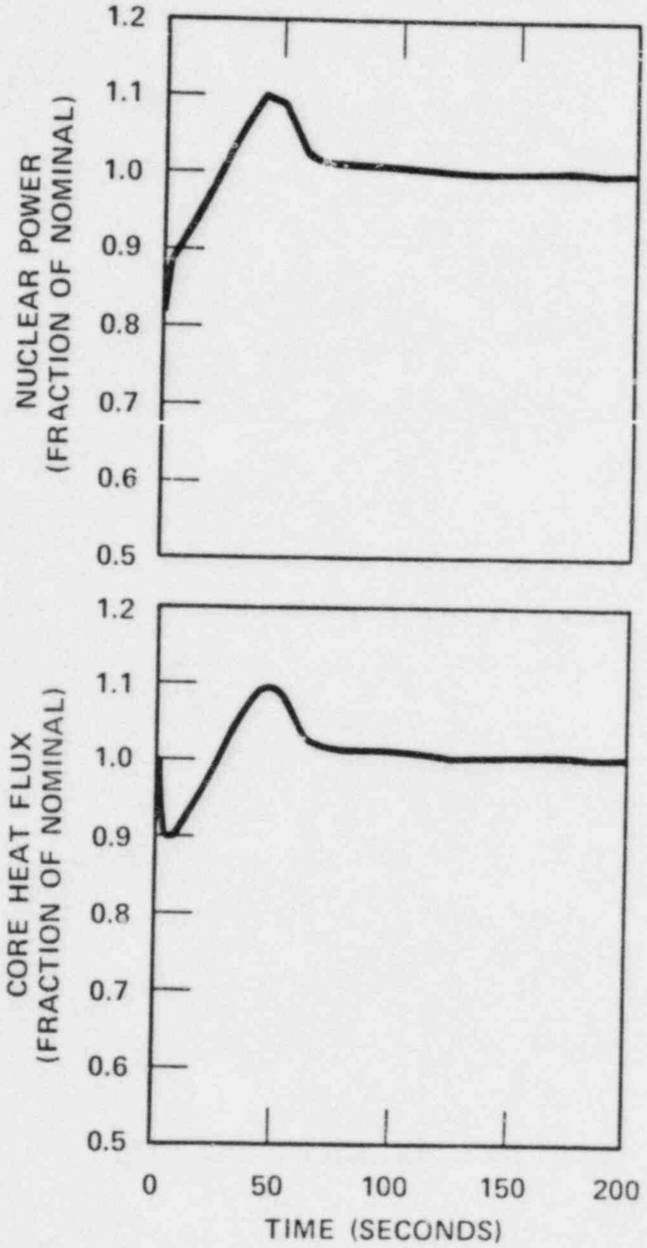
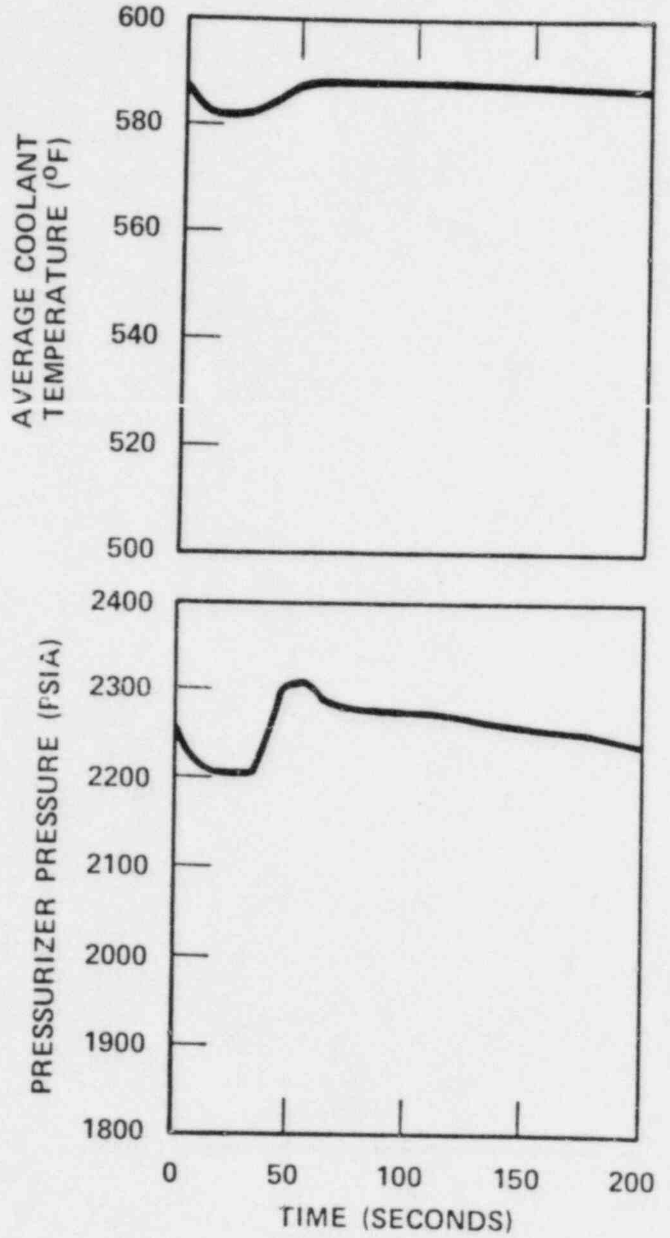


BYRON/BRAIDWOOD STATIONS
FINAL SAFETY ANALYSIS REPORT

Figure 15.4-12a (Sheet 1 of 2)

Typical Transient Response to a Dropped RCCA (or RCCA's) in Automatic Control



BYRON/BRAIDWOOD STATIONS
FINAL SAFETY ANALYSIS REPORT

Figure 15.4-12a (Sheet 2 of 2)

Typical Transient Response to a Dropped RCCA (or RCCA's) in Automatic Control