BWROG Assessment of NRC Information Notice 2007-07

1.0) <u>Summary:</u>

This assessment addresses the condition described by the NRC in NRC Information Notice 2007-07 and in the inspection report referenced therein.

The overall assessment of the condition described in NRC Information Notice 2007-07 by the BWROG is that it represents a condition with a low likelihood of occurrence, with low safety significance and with multiple layers of defense-in-depth currently in place each with the capability to either prevent the condition from occurring or to effectively mitigate the effects of the occurrence without consequence.

It is the position of the BWROG that all BWRs should have a manual operator action tied to their post-fire safe shutdown procedures instructing the operator to implement the requirements of EO-113 should the fire impact the ability to scram. This manual operator action should be endorsed by the NRC for use in both III.G.1 and 2 areas, as well as, III.G.3 and III.L areas. The evaluation provided in this paper and the limited likelihood of occurrence of the condition are considered to be sufficient justification for concluding that this manual operator action is both feasible and reliable.

It is recommended that each BWR review this assessment and assure that their plant specific conditions are consistent with the measures described herein. As a minimum, each licensee should assure that the EOP action to implement the requirements of EO-113 is linked to their post-fire safe shutdown procedures.

2.0) <u>Description of Issue:</u>

NRC Information Notice 2007-07 postulates a condition where two (2) hot shorts could result in the failure of one of four control rods groups to insert during a manual scram from the Control Room. The IN further postulates that with the reactor in this condition the operator rapidly depressurizes the reactor and re-floods the reactor with cold water using a low pressure system. The IN further states:

> "By design, the negative reactivity, added by all four rod groups during a scram, provides adequate shutdown margin to offset the positive void and temperature reactivity [that] would have been added to the vessel [during such a shutdown sequence]".

3.0) <u>Scram System Design Description:</u>

Typically, the Reactor Protection System (RPS) for a BWR consists of two (2) Trip Systems (A and B), each containing two Trip Channels (A1, A2, B1, B2) of sensors and logic. The four channels contain automatic scram logic for the monitored parameters listed below, each of which has at least one input to each of the logic channels:

- Scram Discharge Volume Water Level

- Main Steam Line Isolation Valve Position
- Turbine Stop Valve Position
- Turbine Control Valve Fast Closure
- Reactor Vessel Water Level
- Main Steam line Radiation
- Neutron Monitoring System
- Primary Containment Pressure
- Reactor Vessel Pressure

The RPS automatic trip logic requires at least one channel in each trip system to be tripped in order to cause a scram. This is referred to as one-out-of-two-taken-twice trip logic.

The two RPS Trip Systems are independently powered from their respective RPS Buses. The trip channels (A1, A2, B1, B2) associated with each Trip System (A, B) operate the automatic scram Trip Logic Relays (K14 A-H). The RPS auto scram logic string is sometimes referred to as "trip actuator" or "actuation" logic because the output of the logic is what actually causes the control rods to scram by de-energizing the pilot scram solenoid valves.

The RPS circuits are a fail-safe design in that the circuits are normally energized, and the loss of power, including the loss of offsite power, will initiate the scram.

Once the scram has occurred, re-energization of the RPS logic will not, in and of itself, cause the control rod movement necessary to re-establish reactor criticality.

4.0) Evaluation:

The evaluation performed is divided into two sections. The first section performs a circuit analysis of the scram circuitry. This portion of the evaluation examines the scram circuitry in an effort to determine the set of hot shorts that, should they occur, have the potential to prevent one or more rod groups from inserting. The first section also addresses the significance of the postulated condition and the features currently in place with the capability to prevent or mitigate the effects of the condition. The second section addresses the implications for Appendix R Compliance given the required circuit design for this important safety system and given the potential ramifications of the hot shorts postulated in the first section.

4.1) Circuit Analysis:

Figures 1 through 4 attached to this paper shows portions of the scram circuitry for a typical BWR. Three (3) separate cases involving up to two hot shorts are discussed in this paper.

Case I: (Refer to Figure 1)

Case I attempted to identify the condition described in IN 2007-07. IN 2007-07 concluded that two (2) hot shorts were required to prevent a single rod group from scramming.

The BWROG, however, was unable to identify any circuitry where two (2) fire-induced hot shorts would prevent one of four scram rod groups from inserting.

The BWROG identified that a single hot short in either of the divisionalized trip logics can prevent the scram of a single rod group. This finding is different than the conclusion in IN 2007-07. The finding of the BWROG assessment is a direct consequence of the 1 out of 2 taken twice logic used in the design for the scram function.

The single hot short with the potential for preventing the scramming of a single rod group could occur in either the Trip System A or B Relay Panel. [Refer to Figure 1 attached for a description of the location of the subject hot short, labeled as “Hot Short 1”.] The hot short must occur prior to the operator scramming the reactor. The location of the hot short shown in Figure 1 would be either in one of the Trip System Relay Panels or in a raceway carrying the circuit from the Trip System Relay Panel to the Scram Pilot Solenoid Valves. (Note: For some licensees, the relay panels are located in separate relay rooms outside of the main control room.)

For the hot short in this case to affect the reactivity function, it must remain in effect until such time when the operator depressurizes the reactor and begins re-flooding with a low pressure system. The Emergency Operating Procedures for a BWR instruct the operator not to depressurize the reactor until reactor level reaches the top of active fuel. In a typical BWR, it will take approximately 20 to 25 minutes of boil-off for reactor level to decrease to the top of active fuel. Industry and NRC cable fire testing have shown that hot shorts last for only a few minutes prior to shorting to ground. [EPRI Testing determined the maximum duration of a hot short was 11.3 minutes. CAROLFIRE Testing determined that the maximum duration of a hot short was 7.6 minutes.]

Therefore, it appears unlikely that the required hot short could last for a sufficient amount of time that the impacted control rod group would fail to insert prior to the time when the EOPs directed the operator to depressurize the reactor.

Case II: (Refer to Figure 2)

Case II is one of two cases identified where two (2) fire-induced hot shorts could prevent a full scram. (Note: No conditions were identified where two (2) fire-induced hot shorts were required to prevent a single rod group from scramming.)

Refer to Figure 2 attached for the case where two (2) fire-induced hot shorts could prevent a full scram.

This case postulates a condition where two hot shorts just below the manual scam switches for two trip channels can prevent a full scram. The postulated hot shorts could occur in either the main control room operating bench board or in a raceway carrying the trip circuit to one of the Trip System Relay Panels. The hot short will keep the K15 relays from de-energizing and this will subsequently keep the K14 relays energized. By keeping the K14 relays energized, as shown in Figure 1, none of the rod groups will de-energize and none will insert. Figure 2 shows the location of the two individual hot shorts. One affects the K15B relay and one affects the K15D relay. The K15 relays are de-energized by actuating the manual scram switches in the Control Room on the main control board. Keeping the K15 relays energized by the hot shots shown in Figure 2, will keep the K14 relays energized, as shown in Figures 3. Keeping the K14 relays energized, as shown in Figure 3, will prevent rod group insertion, as shown in Figure 1.

For this case, however, there are numerous other inputs into the scram logic that can override the effects of the hot short affecting the K15 relays. Refer to Figures 3 and 4 for the additional input signals to the scram function. For example, as shown on Figure 4, closure of the MSIVs or reactor level reaching the +13" level will override the effects of the hot shorts affecting the K15 relays and result in a de-energization of the K14 relays and full rod insertion.

Therefore, it appears unlikely that the required hot shorts, even if they were to co-exist, could prevent the scram and cause the reactivity transient described in the IN. This is true because the effect of the hot short would be overriddened by the reduction in reactor level that would be necessary before the operator would take the action to depressurize the reactor prior to making up with a low pressure system.

<u>Case III:</u> (Refer to Figure 3) (Limited to the Trip System Relay Panels)

Case III is similar to Case II. Hot shorts are postulated in the locations shown in Figure 3, the K14 relays will again remain energized. The energization of the K14 relays will prevent the scram for all rod groups.

For this case to occur, the fire must sufficiently damage two separate circuits and the fire induced damage must occur on each circuit simultaneously. Industry and NRC cable fire testing have shown that hot shorts last for only a few minutes prior to shorting to ground. [EPRI Testing determined the maximum duration of a hot short was 11.3 minutes. CAROLFIRE Testing determined that the maximum duration of a hot short was 7.6 minutes.]

Therefore, it appears unlikely that the required hot shorts would co-exist given that the time required for fire damage to the individual cables and fire propagation between relay compartments to occur.

For all of the cases discussed above, regardless of the number of fire-induced hot shorts postulated, the required hot short configuration must occur prior to the operator scramming the unit. For those configurations requiring more than a single hot short, the two hot shorts must exist coincidentally.

The hot short configurations must remain in effect until such time when the operator depressurizes the reactor and begins re-flooding with a low pressure system. The Emergency Operating Procedures for a BWR instruct the operator not to depressurize the reactor until reactor level reaches the top of active fuel.

Additionally, the scenario described in the IN represents a condition more severe than many BWRs would experience due to the availability of additional safe shutdown system capability. Many BWRs also have high pressure systems available for alternative shutdown at their remote shutdown panel. For a BWR with a high pressure system safe shutdown capability, the time available prior to the need to reduce pressure reactor pressure for injection with either a low pressure system or for shutdown cooling would be extended by a number of hours.

Finally, operators for all BWRs are trained on the use of the Emergency Operating Procedures. EO-113 for each BWR provides clear direction to the to either remove RPS power or the vent the SCRAM air header to achieve a full scram.

4.2) <u>Implications for Appendix R Compliance:</u>

For all plants the main operating bench board is in the main control room. At some plants, the relay panels are located in the main control room. In other plants the relay panels are located in a relay room separate from the main control room. For these latter set of plants, some classify the relay room as III.G.3 areas and some classify the relay room as III.G.1 and 2 areas.

This issue, therefore, has implications for redundant safe shutdown under Appendix R Section III.G.1 and 2 and for alternative and dedicated safe shutdown under the requirements of Appendix R Section III.G.3 and III.L.

With respect to Case I, it is clear that none of the methods available under III.G.2 would be effective in preventing the condition. Protection of the subject circuits with a 3 hour fire rated barrier, with a one hour fire rated barrier with automatic suppression and detection or by separation of 20 feet with automatic suppression and detection and no intervening combustibles, would not prevent the occurrence of this event. Additionally, even if the relay panels for each of the four channels are located in separate control/relay room in separate fire areas, the condition could still occur and 3-hour fire rated barriers for each of these postulated fire areas would be ineffective in preventing the occurrence of the condition. The condition postulated in Case I can only be mitigated by the use of a manual operator action consistent with the manual operator actions currently invoked under Emergency Operating Procedure, EO-113.

The conditions described for Cases II and III are similar. Neither of these cases represents a condition that is prevented by the type of redundant train separation invoked under Appendix R, since the postulated hot shorts occur within a single division.

Therefore, the provision of Appendix R cannot be used to address the conditions described in this paper. Re-design of the scram circuitry is not a viable option without compromising the design function of this important safety function. In addition to the features of the RPS system described above, the Alternate Rod Insertion (ARI) system (vents SCRAM air header), Backup Scram Solenoids (vents SCRAM air header), and Standby Liquid Control (SLC) system (inserts sodium pentaborate) provide additional redundant means to achieve reactor shutdown. For areas such as the main Control Room and the Relay Rooms, however, similar fire-induced impacts could be postulated.

This paper has highlighted one example of an area where verbatim compliance with the requirements of Appendix R is insufficient in preventing fire induced damage from potentially impacting safe shutdown. The BWROG believes that this case and, potentially, other like it are the reason why from the initial issuance of Appendix R that certain conditions were considered to be initial boundary conditions for the Appendix R Post-Fire Safe Shutdown Analysis. Assuming that the reactor is scrammed was one of those initial boundary conditions given for the Post-Fire Safe Shutdown Analysis. NRC Generic letter 86-10 in the Response to Question 3.8.4, Control Room Fire Considerations, endorsed the assumption of a reactor trip prior to evacuating the Control Room. Based on this and on the fail-safe nature of the reactor protection system, many licensees assumed and the NRC accepted that a reactor trip was an initial boundary condition for the start of the post-fire safe shutdown analysis, i.e. the plant is scrammed prior to the scram circuitry being damaged by the fire.

Although the BWROG believes that the prior industry position related to the scram is correct and its use provides for a safe plant design, the BWROG also recognizes that fires have some limited potential to impact the scram capability. As a precaution, it is the position of the BWROG that all BWRs should have a manual operator action tied to their post-fire safe shutdown procedures instructing the operator to implement the requirements of EO-113 should the fire impact the ability to scram. This manual operator action should be endorsed by the NRC for use in both III.G.1 and III.G.2 areas, as well as, III.G.3 and III.L areas. The evaluation provided in this paper and the limited likelihood of occurrence of the condition are considered to be sufficient justification for the feasibility and reliability of this manual operator action.

5.0) Risk Assessment:

Given the unlikely set of circumstances required for this condition to occur and to remain in effect until such time that it could pose a beyond design basis concern to the reactor, the risk associated with this issue is judged to be low.

6.0) Safety Assessment:

Given the fact that there are multiple barriers (circuit failure characteristics, design features, procedural guidance and rigorous operator training) in place to prevent the occurrence of this condition, the safety significance of this issue is also judged to be very low.

7.0) Conclusions and Recommendations:

This assessment addresses the condition described by the NRC in NRC Information Notice 2007-07 and in the inspection report referenced therein.

The overall assessment of the condition described in NRC Information Notice 2007-07 by the BWROG is that it represents a condition with a low likelihood of occurrence, with low safety significance and with multiple layers of defense-in-depth currently in place each with the capability to either prevent the condition from occurring or to effectively mitigate the effects of the occurrence without consequence.

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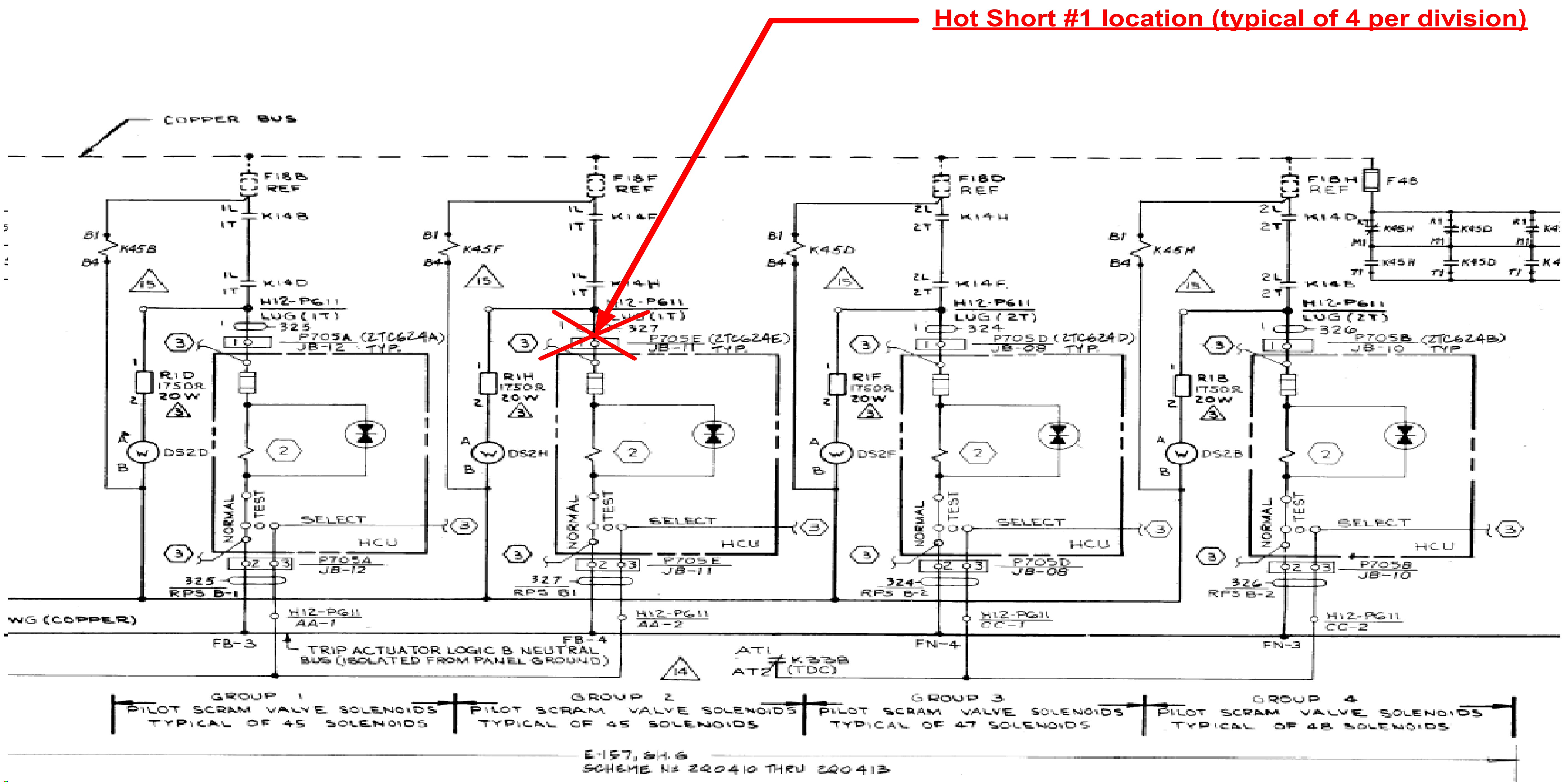
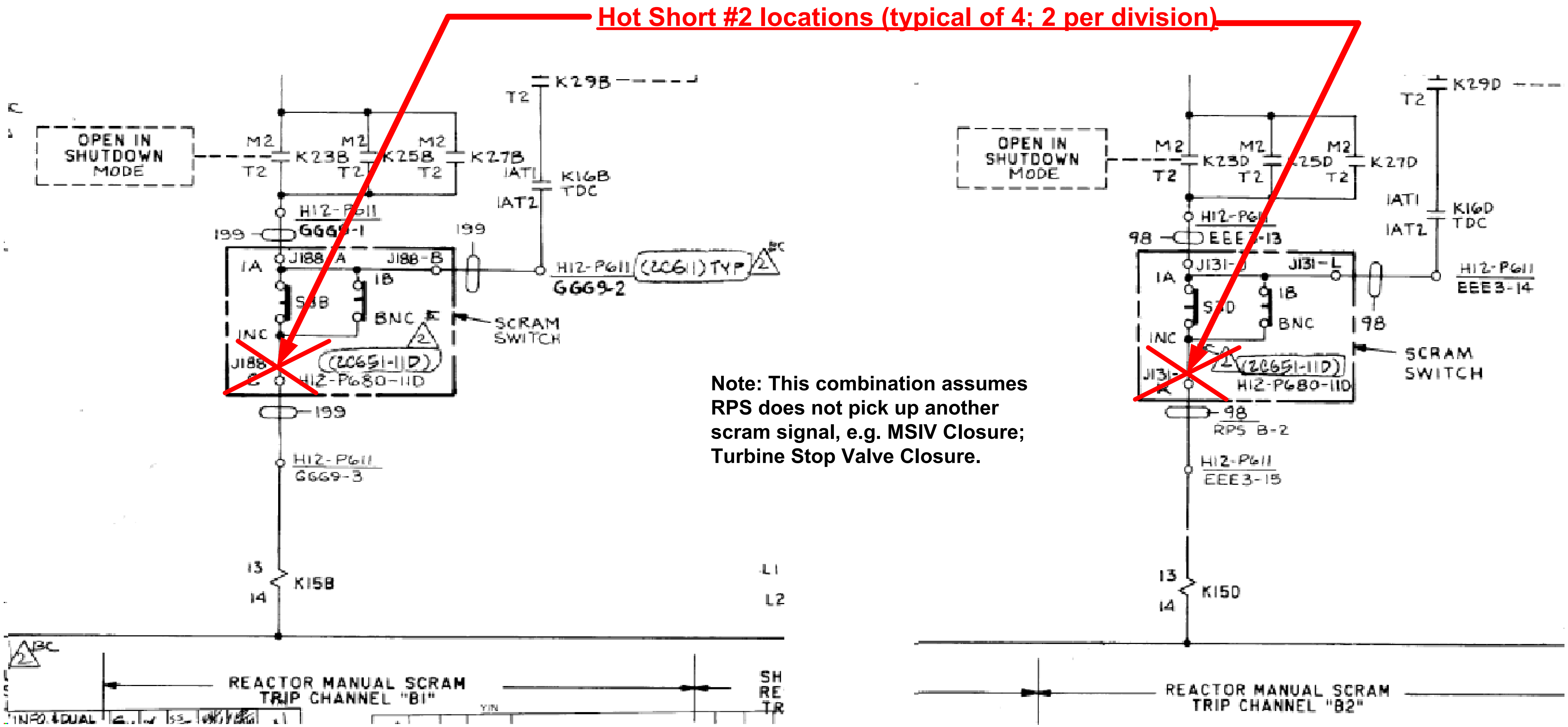


Figure 1 - Relay Panel Circuitry Controlling Individual Rod Groups (typical of two Trip Systems)



Note: This combination assumes RPS does not pick up another scram signal, e.g. MSIV Closure; Turbine Stop Valve Closure.

Figure 2 - Manual Scram Circuitry - Typical of two Trip Systems

Refer to Figure 4 for the remaining set of contacts that affect the automatic scram function

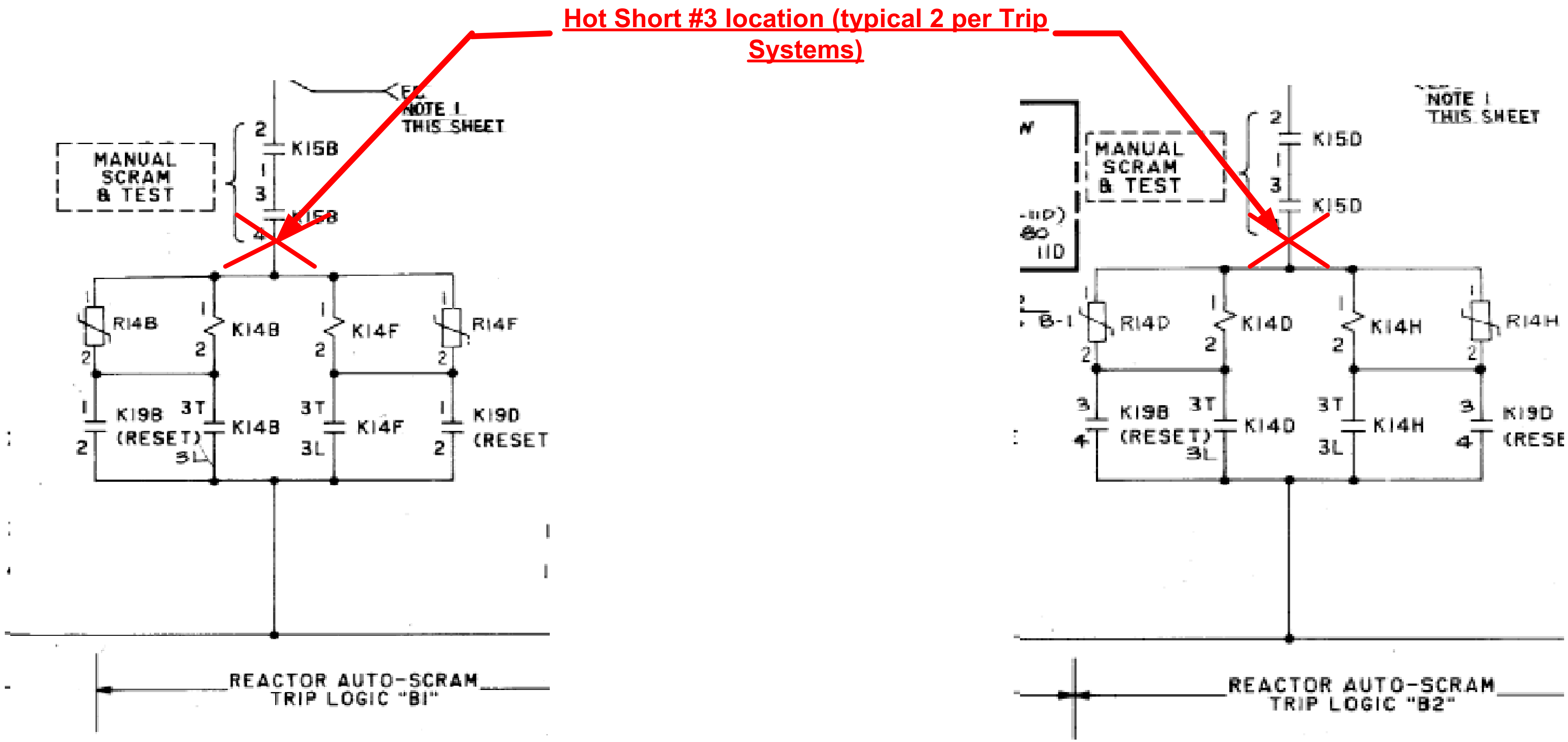
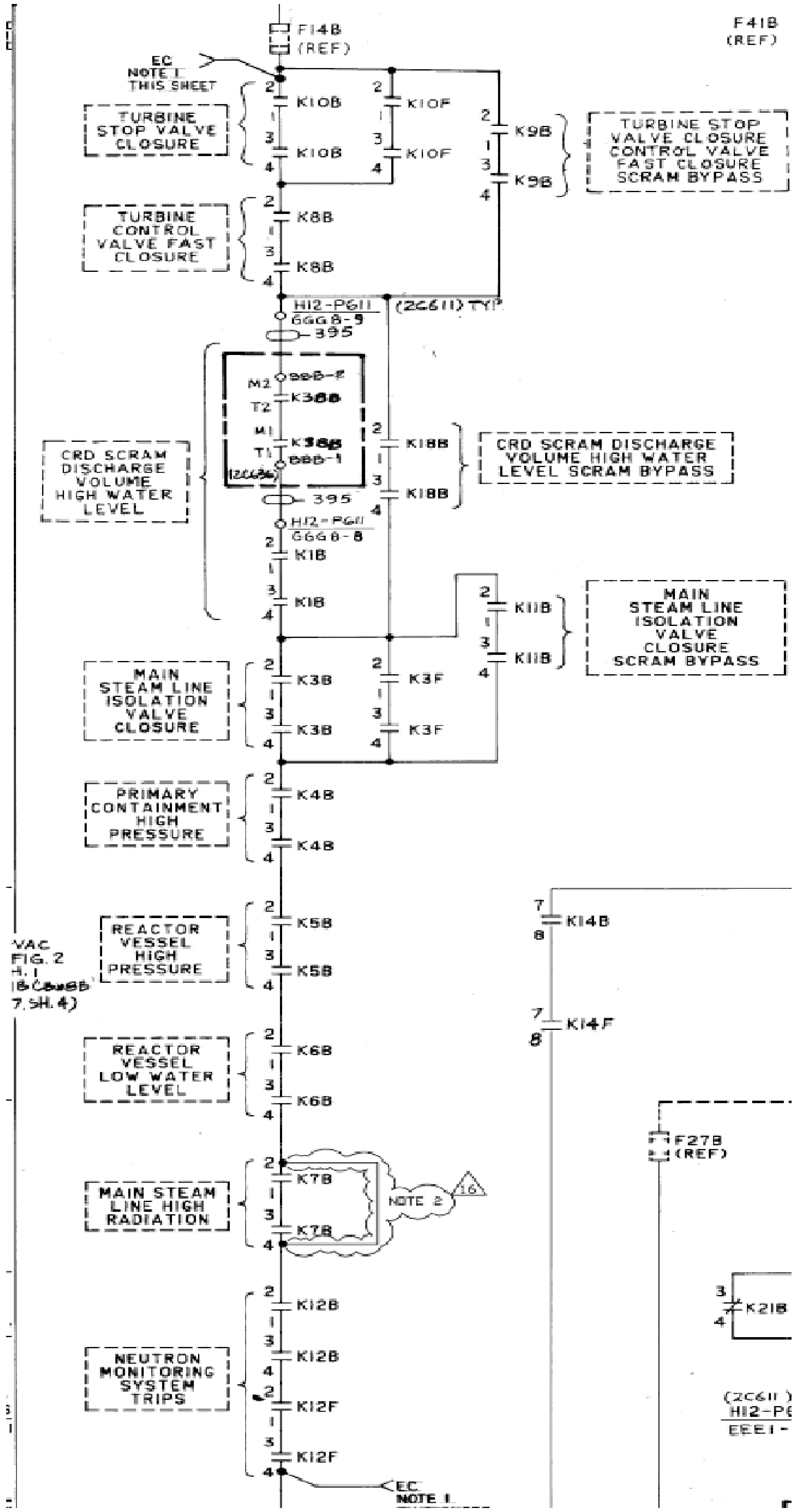


Figure 3 - Reactor Auto-Scram Circuitry - Typical of four Trip Channels in two Trip Systems



<u>**Figure 4- Balance of Auto-Scram Circuitry - (typical of 4 Trip Channels)**</u>