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Transforming Criticality Control Methods for EBR-II Fuel Handling During Reactor Decommissioning

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TRANSFORMING CRITICALITY CONTROL METHODS FOR EBR-II FUEL HANDLING DURING REACTOR DECOMMISSIONING

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1.0 ABSTRACT

A review of the Department of Energy (DOE) request to decommission the Experimental Breeder Reactor-II (EBR-II) was conducted in order to develop a scope of work and analysis method for performing the safety review of the facility. Evaluation of the current national standards, DOE orders, EBR-II nuclear safeguards and criticality control practices showed that a decommissioning policy for maintaining criticality safety during a long term fuel transfer process did not exist. The purpose of this research was to provide a technical basis for transforming the reactor from an instrumentation and measurement controlled system to a system that provides both physical constraint and administrative controls to prevent criticality accidents. Essentially, this was done by modifying the reactor core configuration, reactor operations procedures and system instrumentation to meet the safety practices of ANS-8.1-1983. Subcritical limits were determined by applying established liquid metal reactor methods for both the experimental and computational validations.

2.0 INTRODUCTION

The EBR-II reactor, located on the Idaho National Engineering Laboratory (INEL) site, is the last remaining Liquid Metal Reactor (LMR) in the United States, it is the last LMR power generating facility in this country and it is the only fuel cycle facility capable of effectively reducing the United States High Level reactor waste problem for the long term. In March 1994 the Department of Energy decided to terminate the Actinide Recycle Program[1] and directed a working group to provide redirection of the Argonne National Laboratory (ANL) Nuclear Programs. The recommendation of the working group[2] was to have ANL perform an analysis of engineering issues based on health and safety criteria for this mandated decommissioning effort, and the reactor was shutdown on September 27, 1994.

The primary reason for having to transform the technical practice for ensuring criticality safety is due to the short half-life (e.g. 65 days) of the Sb-Be neutron source relative to the defueling time frame of the core (e.g. years). The long time frame for defueling makes it impossible to comply with the minimum source strength requirements for fuel handling. Thus, by June 1995 the Sb-Be neutron source will have decayed below the 5 counts/second threshold level required by the technical specifications[3]. With a neutron count rate less than 5 counts per second, movement of subassemblies from the reactor core is prevented by low count rate interlocks in the Reactor Safety System (RSS).

For the criticality safety evaluation of this work, the principle focus was to provide a basis for making the transition from relying on instrumentation controls in a reactor to a combination of administrative restrictions and physical restraints. The principle objective was to provide adequate safety when the reactor subassemblies were being moved. The approach for making this transition was to apply the safety practices of ANS 8.1[4] in evaluating the reactor defueling modes. The principle tasks included: evaluating the reactivity of the final core configuration using both computational methods and experiments; implementing new administrative controls in the technical specifications; and providing physical, mechanical and software lock-outs in several reactor systems.

For the modification of the EBR-II reactor's limiting conditions for facility operation (LCO's), an approach similar to the Hanford Fast Flux Test Facility (FFTF) shutdown method[5,6] was applied in order to provide an effective and expedient review process. Essentially, fuel handling for the purposes of reactor defueling would continue as currently defined in the technical specifications, unitl the facility trip and interlock conditions can not be met by the neutron detection system[7]. Once the current technical specifications can no longer be met, then the reactor decommissioning will continue under new defueling technical specifications. One of the principle concerns in modifying the technical specifications is to ensure that the new defueling mode LCO's do not conflict with the remaining technical specifications[8]. The logic of the evaluation process was as follows:

1. The core has been placed in a deep subcritical condition where the k_{eff} is less than 0.95. This was achieved by condensing the core size and removing several high worth subassemblies, in addition to disabling both control and safety rod drives. The evaluation of k_{eff} has been verified both computationally and experimentally.

2. Configuration changes and administrative controls were established in order to prevent additional reactivity from being added to the primary system. This provides the basis for ensuring that criticality can not occur in the reactor once the fuel in the transfer basket is removed from the primary tank.

3. With the first two steps completed, the RSS and Containment Isolation trips related to neutron monitoring will not be necessary. This is due to the fact that these trips are for protection against fuel handling errors which may result in a criticality accident; thus, once a subcritical inventory is achieved in the prescribed manner the system trips can be eliminated.

3.0 SYSTEMS BEFORE MODIFICATION

3.1 Reactor Control Rods

One of the central issues in the decommissioning of the reactor with fuel in the core is the disposition of the control rods. Before the system modifications occur, the control rods provided reactivity by either the adding or removing fissile control rod drivers. However, after the modifications, the control rods will serve solely as a fixed poison in order to ensure a more subcritical system. It should be noted that the EBR-II reactor is fissile fuel controlled, as opposed to poison controlled which is the method used in commercial power reactors.

The control rods were designed to provide reactor power control as well as scram capability in the event of numerous fault event scenerios. A control-rod subassembly is made up of fuel elements with a neutron poison follower above the core. Each control-rod is independently driven into the reactor core by an electromechanical drive mechanism. The drives are identical and independently actuated, and all rods are dropped simultaneously in the event of a reactor scram. The control rod drive mechanism performs essentially two functions. First, it is the connection between the drive and the control rod. Second, it provides the slow speed bi-directional vertical motion for reactor control. All of these operations are combined into a single unit with appropriate interlocks. A diagram of the control rod drive system is shown in figure 1.

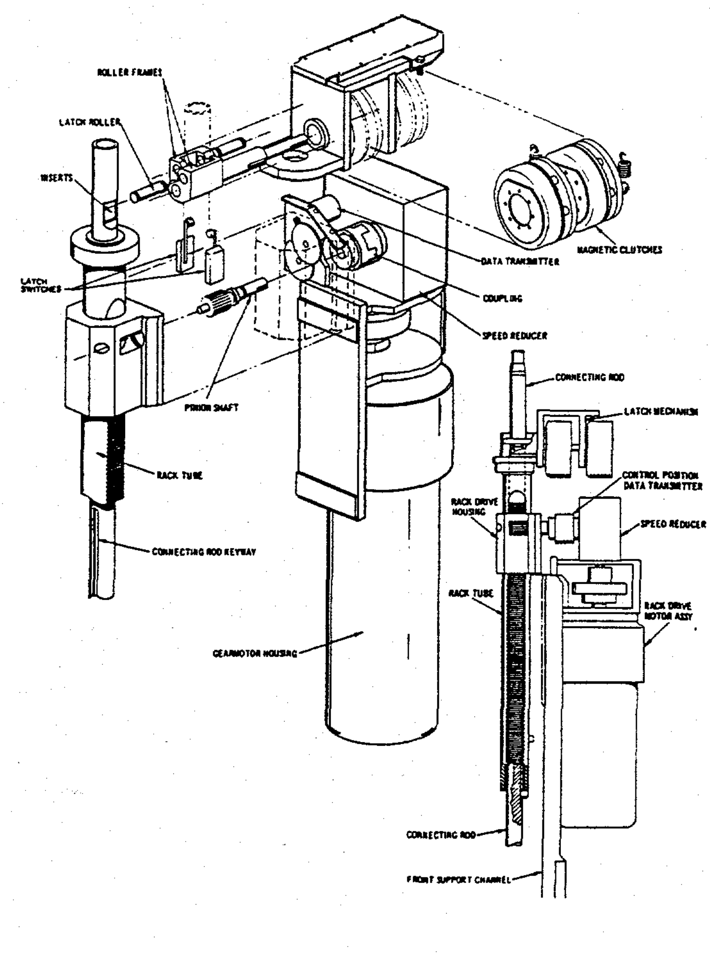

Figure 1. Control Rod Drive Shaft Mechanism.

The control-rod subassembly is disconnected from the control-rod drive mechanism prior to unrestricted fuel handling operations. In the down position, each control-rod drive mechanism actuates the limit switch which is connected to stop the cover lift drive mechanism. The set of circuits are in series during the cover lifting operation, so that in the event that any of the control-rod drive mechanism were inadvertently lifted then power to the cover lifting drive mechanism will be removed to prevent damage to the control-rod drive mechanisms.

3.2 Safety Rods

The two safety rods are connected beneath the reactor to a horizontal I-beam which is connected to two vertical shafts that extend upward through the biological shield. The rods are driven by a synchronous motor and simply raise the system to the "cocked position." When the latch is released the safety rods drop under the force of gravity. A pneumatic shock absorber decelerates the mechanism during the last 5 in. of travel. Fuel-handling requires the safety rods to be in the up position in order to provide a means for decreasing reactivity in the system.

The drive mechanism consists of a gear motor, an electromagnetic clutch, a torsion spring, a gear train coupled to both vertical shafts along with associated instrument and control systems. The safety rods are connected to a horizontal support bar by a bayonet-type lock. The entire system acts as a unit with both rods being dropped simultaneously. A diagram of the safety rod drive system is shown in figure 2.

3.3 Containment Isolation System

The containment building isolation system operates in two modes: partial isolation and full isolation. The partial isolation mode will isolate only the building ventilation system and the fast cover-gas (argon) purge system. This mode is initiated automatically by the radiation monitors on the purge and ventilation exhaust vents. The full isolation mode will isolate the building ventilation system and those lines penetrating the building containment barrier that open into the primary tank, e.g., lines for the Cover Gas Cleanup System, the Experimental Equipment Building, and the fast cover-gas (argon) purge. Full isolation is automatically initiated by earthquake, high flux (power) level, or subassembly outlet temperature. In addition to this, either isolation mode can be accomplished manually. All valves in closed systems (i.e., such as Freon and silicone systems that have a solid barrier between their fluids and the primary-system fluids) that penetrate the building will be isolated only by manual initiation.

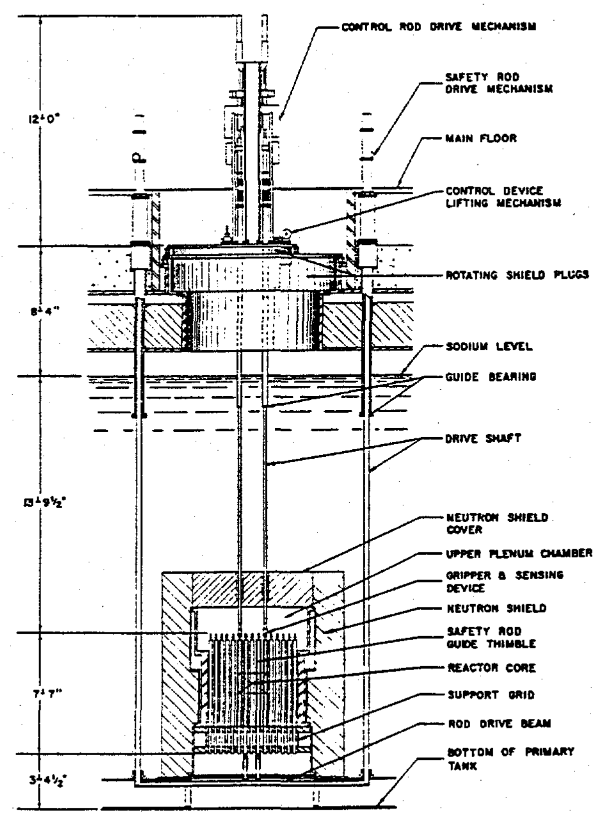

Figure 2. Control and Safety Rod Drive System.

4.0 SYSTEM MODIFICATIONS DESCRIPTION

4.1 Disabling of Reactor Control Rods

In order to eliminate the possibility of reactivity insertion into the reactor core due to either deliberate or accidental rod movement, the control rods will be disabled during the entire time frame of the defueling process. The disabling of the control rods will provide two independent levels of motion impediment which are provided by both electrical and mechanical changes in the current system configuration. The explicit details of the analysis along with the physical changes have been completed under Engineering Change Authorization ECA-5691.

For the electrical changes, the electric power to the scram clutch power supply will be physically disabled so that no control power is available to the system. Essentially, this requires a modification of the Control and Safety Rod Operating Circuits, shown in figure 3. The electrical path for A.C. power to the 90 VDC power supply will be eliminated between the power control relays CP-1 and CP-2.

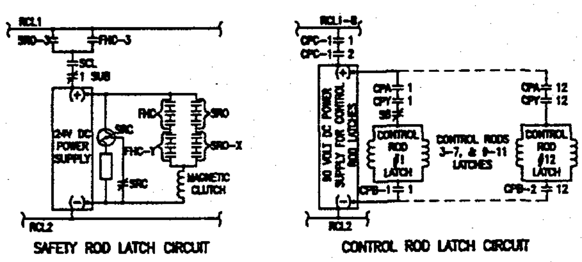

Figure 3. Control and Safety Rod Latch Circuit.

For the mechanical changes, the control rod drive motor will be physically disabled so that the control rod rack tube can not be move. This will be achieved by removing the pinion shaft from the rack tube. This modification is shown in figure 1.

4.2 Fuel Handling with Safety Rods Down

Under the previous version of the technical specifications, unrestricted fuel handling would not have been allowed once the neutron count rate became less than 5 counts per second. This would also be physically prevented by the plant protection system's low level source flux interlocks. However, with the electrical and mechanical impediments provided then reactivity insertion from control and safety rods is not a credible accident scenario. This provides sufficient justification for fuel handling without neutron monitoring. This also provides sufficient justification for removing the monitors from the plant protection system (PPS). This work includes the low level source flux interlocks, the high flux and period trips, and the high flux containment isolation trips.

With the remaining fuel in the reactor configured in a permanent subcritical state it is not necessary to actuate an automatic trip of the reactor. Therefore, the safety rods will be disabled in the down position. This work involves an electrical impediment in the current system configuration. The electrical changes will be similar to those done for the control rods. This work will be completed under Engineering Change Authorization ECA-5694.

4.3 Containment Isolation System Modification

For the Containment Isolation System, three individual containment barriers enclose the EBR-II reactor, each with a separate purpose. The innermost is the primary tank, which holds the reactor core and the primary-sodium coolant. The purpose of this barrier is to limit the escape of activity from the argon cover gas during normal operation of the reactor. The second barrier is the primary containment system. The purpose of this system is to restrict blast damage from a hypothetical reactivity insertion excursion such as the design basis accident. The purpose of this system is to contain the resultant energy release and primary sodium, but not the resultant radioactivity. The third barrier is the reactor containment system. The purpose of this system is to contain any radioactivity which might escape from the primary tank.

The combined set of individual containment barriers are vital to the overall containment of released radiation; however, the original basis for this system analyzed the outer-most barrier. The configuration of the original system for isolation of the reactor building was such that all isolation valves were closed on high sub-assembly outlet temperature, high-reactor building temperature, high reactor building pressure or high radiation level in the building. The containment system was later modified to eliminate the containment high temperature and high pressure trips.

For the Containment Isolation System, the three individual containment barriers that enclose the reactor will remain intact; however, the high linear power level isolation trip will be removed. This modification will be completed under Engineering Change Authorization ECA-5694.

5.0 SAFETY EVALUATION

5.1 Criticality Considerations

The last EBR-II reactor core loading that operated at full power was designated as Run 170A. Using the inventory of fuel in the reactor grid at the shutdown of Run 170A, the fuel loading was re-shuffled into a reference core configuration that (1) had a minimal shutdown excess reactivity at the end of cycle, and that (2) provided the most compact core arrangement possible with the available inventory of subassemblies. This new reference core configuration was designated as Run 170B. The purpose of this rearrangement of subassemblies was to guaranteed that in the grid no significant increase in reactivity could occur. Furthermore, with the additional removal of some high neutron importance fuel from the grid, the reactor core would be sufficiently subcritical. This Run 170B core configuration was brought critical on September 30, 1994; and the configuration of the safety and control rods is shown in table 1.

Table 1. Run 170B Critical Configuration

Rod Type	Position, cm
Safety Rods Bank	35.56
Control Rod Bank	35.56
Control Rod No. 10	23.14

Applying the control rod calibration, the measured zero-power excess reactivity was 27 cents. Thus, the Run 170B core loading did not have sufficient reactivity to return to full power with all rods inserted, since approximately 32 cents of excess reactivity is required to get EBR-II to full power. Furthermore, the measured control rod reactivity for the shutdown configuration[9] with safety and control rods down was –7.57 dollars. The purpose of the second re-loading objective was to ensure that any further rearrangement of subassemblies in the core would not add significant reactivity to the system. A special rod drop measurement was made during the startup to determine the relationship between neutron detector count rates and the reactor subcriticality. This calibration measurement allowed the experimental determination of all subsequent subcritical core configurations using the modified source multiplication (MSM) method. The MSM experimental subcriticality[9] was –7.7 dollars and the Dif-3D calculated subcriticality was –8.2 dollars.

In order to reach a subcritical eigenvalue less than 0.95, the core loading was modified slightly in the first reactor defueling loading, RDF-1, in order to establish a shutdown configuration that was significantly subcritical. Two high neutron importance fuel-bearing experimental subassemblies that were in the reactor core for run 170B were removed and replaced with steel subassemblies. The final shutdown configuration of this loading had an experimentally measured subcritical reactivity of –10.2 dollars, and the calculated value was –10.9 dollars. This negativity reactivity corresponded to an experimental eigenvalue measurement of 0.935 ± 0.008. The detailed calculational and experimental analyses supporting these results are provided in the references[9,10]. It should be further noted that even for the shutdown and fuel handling configuration, with the safety rods in, the eigenvalue is 0.945. Thus the criticality safety analysis does not have to take into account the safety rods in order to maintain the established 0.95 requirement.

All subsequent reactor defueling loadings are certain to be more subcritical than the reference case 170B. In terms of procedure, all driver fuel and experimental subassemblies will be removed from the basket and reactor tank before any more fuel handling is done in the core region. This will preclude adding fuel to the core even in the unlikely event of a loading error. Having the core subassemblies already in nearly the most reactive arrangement implies that any accidental rearrangement of the core cannot add significant reactivity. Second, The potential for adding reactivity by inserting control or safety rods will be eliminated by disabling their drives electrically and mechanically. The High Worth Control Rod subassemblies will not be removed from the core until after all other fuel assemblies have been removed from the reactor. Furthermore, the blanket is such a low neutron importance region that the loading changes from blanket material to stainless steel are negligible[11].

The RDF-1 configuration is shown in figure 4. The configuration resulted from removing fuel from grid positions 3A2 and 4A1 (after run 170B) as follows:

Table 2. RDF-1 Fuel to Stainless Steel.

Grid Position	S/A Out	S/A In
3A2	X521	K031
4A1	X492B	K030

The grid positions listed above are the only locations where reactivity can be added by physical rearrangement of the subassemblies in the core. Also, the only possible rearrangements of subassemblies are:

1. Move a half-driver from a 5N2,4 location to 4A1 and K030 to row 5. This results in a reactivity increase of 15 cents.

2. Move a half-driver from 5N2,4 location to 3A2 and K031 to row 5. This results in a reactivity increase of 30 cents.

3. Move a driver from 4N2,3 location to 3A2 and K031 to row 4. This results in a reactivity increase of 34 cents.

These three transfer schemes represent the only possible ways for increasing reactivity because row 5 does not contain any full-worth drivers, and all other fuel movements are prevented by mechanical interlocks in the subassembly design. Thus, if all fissile material is removed from the storage basket, the largest possible reactivity increase is 49 cents if all administrative controls fail and both operations 1 and 3 were to occur simultaneously.

With the driver fuel and experimental subassemblies removed from the fuel transfer basket and primary tank, then the possibility of a criticality is eliminated and all technical specifications associated with the criticality control of the reactor can be eliminated from the LCO's. This includes the low level source flux interlocks, the high flux and period trips, and the high flux containment isolation trips. With the removal of these system the reactor criticality control will then be completely transformed from instrument and measurement means to physical constraint and administrative means.

It should be re-emphasized that this transformation of criticality control methods is based on two contingencies. First, all driver fuel and experimental subassemblies in the fuel transfer basket must be removed from the primary tank. Furthermore, this work must be accomplished before unrestricted fuel handling would be allowed using TSR's which allow fuel handling with the safety rods down and no neutron source requirements. Second, the fuel must be completely removed from the primary system before any sodium removal takes place. This requirement eliminates the necessity for any water reflood evaluation of the core configuration.

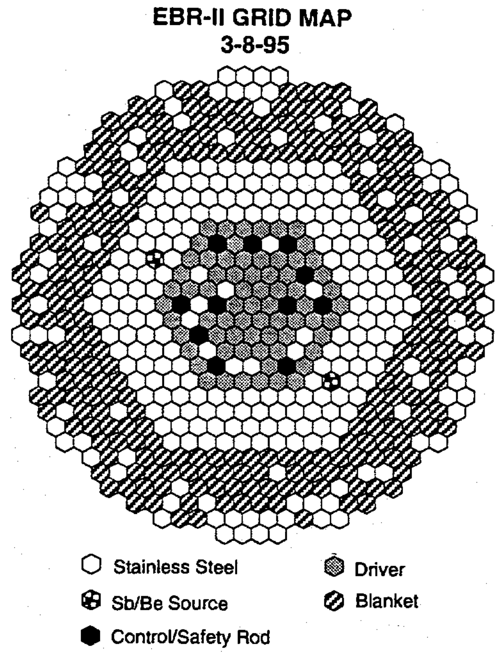

Figure 4. EBR-II Grid Map for RDF-1 Loading.

6.0 CONCLUSIONS

This safety evaluation established the core loading following RDF-1 to be deep subcritical with an eigenvalue of 0.935. Also, inserting all control rods and safety rods into the core will not take the reactor critical with this core configuration. The modifications discussed in section 3 render the control rods and safety rods inoperable, thus there is no way to move the control rods and safety rods to the fully inserted positions. The possibility of making the system critical is eliminated and all technical specifications associated with the criticality control of the reactor unrestricted and restricted fuel handling can be eliminated from the limiting conditions for defueling operations.

7.0 ACKNOWLEDGMENTS

This work represents a composite effort of a number of people on staff at ANL-west including: D. Cutforth, L. Hansen, C. Matthews, G. McAffee, R. Schaefer, and W. Schorzman.

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