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10 CFR 50.54(q)

RS-10-043
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ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Response to Request for Additional Information Related to Request for NRC Approval of Changes to the Clinton Power Station Emergency Plan (TAC No. ME1727)

Reference: Letter from J. L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "Request for NRC Approval of Changes to the Clinton Power Station Emergency Plan," dated June 19, 2009

In the referenced letter, Exelon Generation Company, LLC (EGC) requested a change to the Clinton Power Station, Unit 1 (CPS) Emergency Plan. The proposed change requests a revision to Table B-1, "Minimum Staffing Requirements for the On-Shift Clinton Station ERO," to increase the Non-Licensed Operator staffing from two to four, allow in-plant protective actions to be performed by personnel assigned other functions, and replace a Mechanical Maintenance person with a Non-Licensed Operator.

During the NRC's review of the Reference document, the NRC found that additional information was required to support its review. The requested information is provided in the attachment to this letter.

There are no regulatory commitments contained within this letter. Should you have any questions concerning this letter, or require additional information, please contact Mitchel Mathews at (630) 657-2819.

Respectfully,

Jeffrey L. Hansen
Manager – Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1. Response to NRC Request for Additional Information
2. Review of the CPS/USAR Appendix 15A Design Basis (Postulated) Accidents

Request No: 1 Applicable Submittal Section: III.a

Are radiation work permits available / active for use during emergency conditions, i.e., higher set points for dose and dose rate? If not, what is the timeframe and process to develop/activate for workers to use?

Response:

Radiation work permits (RWPs) are in place, and active for use with set points for dose and dose rate that reflect expected conditions. In an emergency, when expected conditions would likely change, electronic dosimeter set points can be adjusted via a network computer in the Radiolocially Controlled Area (RCA) access control program once the on-shift Radiation Protection Technicians (RPTs) have determined actual dose rates. Onsite Radiation Protection (RP) personnel determined that this routine evolution can be performed within approximately three minutes by the on-shift RPTs. If dose rates are unknown, continuous RPT assistance will be required. Additionally, RPTs can modify the setpoints on electronic dosimetry that is maintained in the fast activation mode.

Request No: 2 Applicable Submittal Section: III.a

Section III.a states, "Operations maintain high radiation area keys for needed access under emergency conditions."

1. *Are high radiation area briefings procedurally required to inform Operation personnel of area dose rates prior to entry for mitigating actions?*
2. *Are Operations personnel qualified to perform self-monitoring in areas where radiation levels have not been identified?*

Response:

In accordance with EGC procedure RP-AA-460, "Controls for High and Locked High Radiation Areas," Sections 4.3 and 4.4, an individual entering a High Radiation Area (HRA) or a Locked High Radiation Area (LHRA), is required to receive a briefing in accordance with RP-AA-460, Attachment 5, and must be equipped with one or more of the following:

- A radiation monitoring device which continually indicates the radiation dose rate in the area
- A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a pre-set integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel made aware of them.
- A Radiation Protection (RP) individual qualified in RP procedures with a radiation dose rate device, who is responsible for positive control over the activities in the area and shall perform periodic radiation surveillances at the frequencies specified in the RWP.

One of the emergency procedures for the Shift Manager, EP-AA-112-100-F-01, "Shift Emergency Director Checklist," requires contacting RP or the Operations Support Center (OSC)

and requesting that an RPT be assigned to the Control Room (CR) to support ongoing CR Emergency Response Team activities.

During emergency conditions when personnel must enter HRAs to perform immediate actions, they are not required to be informed of area dose rates prior to entry as long as they are provided a qualified RP individual to maintain positive control over the activities. This coverage is provided by the on-shift complement of RPTs. Operations personnel are not qualified to perform self-monitoring in these areas.

Request No: 3 Applicable Submittal Section: III.c

Section III.c states, "Emergency issuance requires a TLD number and name of the person to who it is issued. This task does not require an ANSI qualified RPT to Perform." Are on-shift ERO members trained / knowledgeable of how to perform this task?

Response:

There is no specific training required for issuing replacement TLDs in an emergency situation. Not all on-shift Emergency Response Organization (ERO) members are knowledgeable of how to perform this task; however, the on-shift complement of RPTs are knowledgeable of the location of replacement TLDs and if required in an emergency, will issue replacement TLDs to the remainder of the on-shift ERO members.

Request No: 4 Applicable Submittal Section: III.c

Section III.c states, "If an electronic dosimeter is lost or damaged, additional electronic dosimeters are maintained in a fast-activation mode for immediate monitoring. This task does not require an ANSI qualified RPT to Perform." Are on-shift ERO members trained / knowledgeable of how to perform this task?

Response:

The on-shift complement of RPTs that are ERO members are trained and knowledgeable on the location of the electronic dosimeters that are maintained in the fast-activation mode and the procedure for activating them. These personnel are responsible for assisting all on-shift ERO members in performing this tasks during an emergency.

Request No: 5 Applicable Submittal Section: III.c

Section III.c states, "Additionally, if an electronic dosimeter is lost or damaged, self-reading pencil dosimeters are pre-staged for emergency use. Issuance of these do not require an ANSI qualified RPT." Are onshift ERO members trained / knowledgeable of how to perform this task?

Response:

There is no specific training required to allow personnel to issue self-reading pencil dosimeters; however, not all on-shift ERO members are required to be knowledgeable in this process. All on-shift RPTs that are designated as ERO responders are knowledgeable of the location and issuance procedures for self-reading pencil dosimeters that are pre-staged for use during an emergency, and can issue them to the remaining on-shift ERO members as necessary.

Request No: 6 Applicable Submittal Section: III.d

Section III.d states, "Dose rates in the auxiliary building, fuel building and containment would make these areas inaccessible one hour following an event based on this analysis. The dose rates from this table indicate that the majority of the plant areas needed for access to perform mitigating actions remain accessible without RP coverage following an event." Do plant procedures require access to the auxiliary building, fuel building or containment to perform mitigating actions following an event? If so, would these areas be accessible without RP coverage?

Response:

Electronic access control is used by on-shift RPTs to limit any access to the RCA while specific area boundaries are being established. No personnel access to containment would be required for any event.

Auxiliary Building access may be required for motor control center (MCC) breaker operation as a contingency action during certain events. If required, on-shift RPTs will provide continuous coverage such as dose rate surveys and access control monitoring should event conditions necessitate an entry into the Auxiliary Building.

Fuel Building access would not be required for operational event mitigation. Moreover, a Spent Fuel Pool accident would likely occur during spent fuel movement, when a dedicated RPT is provided to support the activity. This activity is performed with RPT staffing levels that are above normal shift staffing.

Request No: 7 Applicable Submittal Section: III.e

Section III.e states, "A review of the design basis events verifies that two RP personnel can perform all the RP tasks necessary to respond to the event for the initial 60 minutes." Please provide a copy of this review.

Response:

The Nuclear Safety Operational Analysis (NSOA) which is Appendix A attached to the CPS USAR Chapter 15, "Accident Analyses," is a comprehensive, total plant, system-level, qualitative Failure Modes and Effects Analysis, relative to all the Chapter 15 events considered on a pre-Alternative Source Term basis. These analyses include protective sequences utilized to accommodate the events and their effects, and the systems involved in the protective actions.

Safety related functions, structures and systems are a subset of the safety functions, structures and systems identified in the NSOA. It captured generic developments encompassing the design, calculation, testing and operating experience of the early BWR product lines. The NSOA identifies on a generic system level basis, those systems which should be the subject of technical specifications, and the safety systems utilized during the different modes of plant operation.

One of the main objectives of the NSOA is to identify all essential protection sequences and to establish the detailed equipment conditions essential to satisfying the nuclear safety operational criteria. The spectrum of events examined in Chapter 15 represents a complete set of plant safety considerations. The main objective of the earlier analyses of Chapter 15, is to provide detailed "worst case" (limiting or envelope) analyses of the plant events. The "worst cases" are correspondingly analyzed and treated likewise in this appendix but in light of frequency of occurrence, unacceptable consequences, assumption categories, etc.

The evaluation used in determining the RP coverage for repair, corrective actions, search and rescue, first-aid, and firefighting discussed in Section III.e of the submittal reviewed the Chapter 15 events (including those discussed in the NSOA), to determine which events would require RP technician support. The results of this review were documented in Attachment 4 of the evaluation that was prepared for the proposed changes in accordance with 10 CFR 50.54(q), and is provided as Attachment 2. Throughout Attachment 2, reference is made to figures and tables (e.g., Table 15A.6-4). The tables and figures referenced in Attachment 2 are found in Appendix 15A of the CPS USAR.

Since no on-shift RPT actions are required by a control rod drop accident, it was not discussed in the evaluation that was performed in accordance with 10 CFR 50.54(q). Similarly, due to additional RPTs beyond the normal shift complement being provided to support fuel handling activities during refueling outages, and the minimal RPT actions required for fuel handling accidents that might occur during operation at power, fuel handling accidents were not discussed in the evaluation that was performed in accordance with 10 CFR 50.54(q). To ensure any questions that may arise regarding these events are addressed, an evaluation of these events is included below.

Description of Event 40 - Control Rod Drop Accident (CRDA)

The control rod drop accident (CRDA) results from an assumed failure of the control rod-to-drive mechanism coupling after the control rod (very reactive rod) becomes stuck in its fully inserted position. It is assumed that the control rod drive is then fully withdrawn before the stuck rod falls out of the core. The control rod velocity limiter, an engineered safeguard, limits the control rod drop velocity. The resultant radioactive material release is maintained far below the guideline values of 10 CFR 100.

The control rod drop accident is applicable only in operating State D as defined in CPS USAR Table 15A.3-1. The control rod drop accident cannot occur in State B because rod coupling integrity is checked on each rod to be withdrawn if more than one rod is to be withdrawn. No safety actions are required in States A or C where the

plant is in a shutdown state by more than the reactivity worth of one rod prior to the accident.

Table 1: CPS USAR Table 15A.3-1

Conditions	States			
	A	B	C	D
Reactor vessel head off	X	X		
Reactor vessel head on			X	X
Shutdown	X		X	
Not shutdown		X		X

CPS USAR Figure 15A.6-40 presents the different protection sequences for the control rod drop accident. As shown in Figure 15A.6-40, the reactor is automatically scrammed and isolated. For all design basis cases, the neutron monitoring, reactor protection, and control rod drive systems will provide a scram from high neutron flux. Any high radiation in the containment areas will initiate closure of other possible pathways to atmosphere, as necessary.

After the reactor has been scrammed and isolated, the pressure relief system allows the steam (produced by decay heat) to be directed to the suppression pool. Initial core cooling is accomplished by either the RCIC or the HPCS or the normal feedwater system. With prolonged isolation, as indicated in Figure 15A.6-40, the reactor operator initiates the RHRS/suppression pool cooling mode and depressurizes the vessel with the manual mode of the ADS or via normal manual relief valve operation. The LPCI, LPCS and HPCS maintain the vessel water level and accomplish extended core cooling. Isolation of turbine-condenser fission product releases will also be maintained.

Event 40 Evaluation

The dropping of a control rod results in a high reactivity in a small region of the core. This would result in a highly peaked power distribution with the possibility of fuel damage. The Local Power Range Monitors and Average Power Range Monitors would register the rapid increase in power and a scram may result. The Rod Control & Information System limits the worth of any control rod which could be dropped by regulating the withdrawals sequence. This system prevents the movement of an out-of-sequence rod up to the low power setpoint (LPSP). The termination of this event is accomplished by automatic safety features of inherent shutdown mechanisms; therefore, no operator action during the event is required.

Main Steam Line high radiation and/or Off-Gas (OG) activity may increase should core damage occur, which could in turn, result in automatic Off-Gas system bypass valve closure due to high OG activity, or automatic OG post treatment isolation on a high radiation condition.

Since the systems discussed respond automatically, without local operator action, no RPT support actions are required for this event.

Description of Event 41 - Fuel Handling Accident

Because a fuel-handling accident can potentially occur any time when fuel assemblies are being manipulated, either over the reactor core or in a spent fuel pool, this accident is considered in all operating states. Considerations include mechanical fuel damage caused by drop impact and a subsequent release of fission products. The protection sequences pertinent to this accident are shown in Figure 15A.6-41. Containment and/or auxiliary fuel building isolation and standby gas treatment operation are automatically initiated by the respective ventilation radiation monitoring systems.

Event 41 Evaluation

The radiological coverage required for this event during refueling outages would be provided by the augmentation RPTs assigned to the work activity. These RPTs are above the normal on-shift complement. Actions would include radiation and airborne surveys around the affected pool area. Personnel would be evacuated from the immediate area and recovery plans would be developed for the event. There are no routine operator actions in the fuel building that would require access to the pool areas during power operation; therefore, impact to on-shift RP resources for fuel handling events that occur during operations at power would be minimal.

To aid in review of the on-shift RPT actions required for the remaining NSOA events discussed in the evaluation performed in accordance with 10 CFR 50.54(q), the evaluations of these events is provided below.

Event 42 (i.e., Loss-of-Coolant Accidents (LOCAs) Resulting from Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary Inside Containment) - Evaluation

The actions described for NSOA Event 42 are CR actions. If local manual valve isolation is required, an operator would be dispatched to the Auxiliary Building MCC cabinets to open containment isolation breakers. RPT support would be limited to verification of dose rates and stay times in the Auxiliary Building. There is no breech of a fluid systems associated with this activity.

Event 43, 44, and 45 (i.e., LOCAs Resulting from Postulated Piping Breaks Outside Containment) - Evaluation

All actions required for these accidents are performed in the CR with no RPT support required.

Event 46 (i.e., Gaseous Radwaste System Leak or Failure) - Evaluation

All actions required for this accident are performed in the CR with no RPT support required.

Event 47 (i.e., Augmented Offgas Treatment System Failure) - Evaluation

RPT support could be required for air sampling and area dose rates in the area if an operator were to be dispatched. With steam line isolation, the need to take any actions outside of the CR for this event would be eliminated and therefore, no RPT support would be required.

Event 48 (i.e., Liquid Radwaste System Leak or Failure) - Evaluation

RPT support would be required for establishing boundaries for the effected areas. Isolation of systems and tanks is performed from the Radwaste Control Room, which limits the need for operator actions in the affected areas, and minimizes the time required by an RPT to provide worker monitoring.

Event 49 (i.e., Liquid Radwaste System - Storage Tank Failure) - Evaluation
RPT support would be limited to control of areas affected by the event. Radwaste tank rooms are currently locked areas due to operational radiation levels. Therefore, no additional RPT support would be required for tank failures.

In summary, all NSOA events discussed in CPS USAR, Appendix 15A were evaluated to ensure that two on-shift RPTs can provide an adequate response to these events. This evaluation concluded that two RPTs is an appropriate on-shift complement.

Request No: 8 Applicable Submittal Section: Page 7 of 11

The discussion for "Replacing Mechanical Responder with a Nonlicensed Operator (NLO)," states, "Overall, for all items reviewed the need for maintenance personnel within the first 90 minutes of an emergency condition is limited to those actions associated with the Emergency Operating Procedures or for troubleshooting or abnormal system alignment to operated equipment that did not respond as expected during the event."

NUREG-0654/FEMA-REP-1, Table B-1 guidance for repair and corrective actions states that two individuals, one with mechanical maintenance / rad waste operator experience and one with electrical maintenance / I&C experience, should be designated on-shift, but may be provided by shift personnel assigned other functions.

The staff views that the function of the maintenance position during the time frame in question is to provide for minor or limited scope damage repair and corrective functions such as:

- *Identification and operation of faulty valves, clogged filters, packing and/or seal adjustments, install/remove hoses for draining and venting of plant piping and equipment, and/or troubleshooting.*

Are NLO's trained / qualified to perform minor or limited scope damage repair and corrective actions?

Response:

Identification and operation of faulty valves

Non-Licensed Operators (NLOs) are trained to recognize indications of faulty valves such as an overload condition on a motor operator, sticking or otherwise bound valve stems, or other forms of mechanical degradation such as through-wall body leakage. Additionally, NLOs are trained and qualified in the mechanical manual override feature of motor-operated valves, and Procedure OP-CL-108-101-1001, "General Equipment Operating Requirements," Section 3.14, "Disabling Air Operated Valves," permits operators to disable vane-type air operated valves. These air-operated valves may include, but are not limited to, tank inlets, and primary and secondary recirculation valves.

Identification and operation of clogged filters

Non-Licensed Operators are trained and qualified to recognize indications of a clogged filter or strainer such as high differential pressure and/or reduced flow. Moreover, operators are trained to perform the system manipulations that are required to shift to the standby filter when indications of clogged filters or strainers on duplex-type units exist, thus permitting normal system operation.

Operators do not perform maintenance on clogged, non-duplex filters; however, operators can perform the system manipulations and safety tagging necessary to isolate and clear the affected filter or strainer for work. This work can be performed in preparation for the arrival of the maintenance support required to replace the clogged filter.

Identification and operation of issues requiring packing and/or seal adjustments

Non-Licensed Operators are trained and qualified to recognize packing and other forms of mechanical seal leakage, and can perform the system manipulations required to isolate and tagout the affected component in preparation for the arrival of the maintenance support required to effect repairs.

Installation and removal of hoses for draining and venting of plant piping and equipment

The Non-Licensed Operators are trained to install and remove hoses for the purposes of draining and venting plant piping and equipment.

Troubleshooting

As for the identification and/or troubleshooting of mechanical equipment problems, it is an expectation that all indications of the conditions are recorded on an Issue Report, with recommendations for the actions necessary to return to component to its normal condition.

In summary, as discussed above, NLOs are trained and qualified to perform the tasks necessary to effect limited scope damage repairs and corrective actions.

Request No: 9 Applicable Submittal Section: Page 9 of 11

Final paragraph states, "Additionally, the station has successfully demonstrated the capability to fully staff and activate the ERO facilities in a September 16, 2004 off-hours augmentation drive-in drill." Is this the only time that this off-hours augmentation has been performed and are there any scheduled to periodically validate ERO response capabilities in the future?

Response:

In accordance with EP-AA-1000, "Exelon Nuclear Standardized Radiological Emergency Plan," Section N.2.f, *"At least once per drill cycle (every 6 years), an off-hours, unannounced activation of the ERO Notification System with actual response to the emergency facilities is conducted by each station."* This activity also meets the NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Section N.1.b, requirement to conduct an off-hours, unannounced drill once per cycle.

The September 16, 2004, drill is the latest drill that has been conducted. The previous drill was conducted in 1998 in accordance with the six-year cycle. The next drill is scheduled in 2010 in accordance with the requirement.

Attachment 2

10 CFR 50.54(q) Evaluation #09-54, Attachment 4,

Review of the Clinton Power Station Updated Safety Analysis Report, Appendix 15A Design Basis (Postulated) Accidents

ATTACHMENT 4

Review of the CPS/USAR appendix 15A Design Basis (Postulated) Accidents

Design Basis Accident (DBA) is a hypothesized accident the characteristics and consequences of which are utilized in the design of those systems and components pertinent to the preservation of radioactive material barriers and the restriction of radioactive material release from the barriers. The potential radiation exposures resulting from a design basis accident are greater than for any similar accident postulated from the same general accident assumptions. Specific events are described in Table 15A.6-4.

15A.3.3.4 Design Basis Accidents

Accidents are defined as hypothesized events that affect the radioactive material barriers and are not expected during plant operations. These are plant events, equipment failures, combinations of initial conditions which are of extremely low probability (once in 100 years to once in 10,000 years). The postulated accident types considered are as follows:

(1) Mechanical failure of a single component leading to the release of radioactive material from one or more barriers. The components referred to here are not those that act as radioactive material barriers. Examples of mechanical failure are breakage of the coupling between a control rod drive and the control rod.

(2) Arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the reactor coolant pressure boundary. This kind of accident is considered only under conditions in which the nuclear system is pressurized.

For purposes of analysis, accidents are categorized as those events that result in releasing radioactive material:

(1) From the fuel with the reactor coolant pressure boundary, reactor building and auxiliary building initially intact. (Event 40)

(2) Directly to the containment. (Event 42)

(3) Directly to the reactor, auxiliary, or turbine buildings with the containment initially intact. (Events 40, 43, 44, 45, 50)

(4) Directly to the reactor or auxiliary buildings with the containment not intact. (Events 41, 50)

(5) Directly to the spent fuel containing facilities. (Events 41, 50)

(6) Directly to the turbine building (Events 46, 47)

(7) Directly to the environs (Events 48, 49)

The effects of various accident types are investigated, with consideration for the full spectrum of plant conditions, to examine events that result in the release of radioactive material.

15A.6.2.3.1 Radioactive Material Release Control

Radioactive materials may be released to the environs in any operating state; therefore, radioactive material release control is required in all operating states. Because of the significance of preventing excessive release of radioactive materials to the environs, this is the only safety action for which monitoring systems are explicitly shown. The offgas vent radiation monitoring system provides indication for gaseous release from the offgas system. Gaseous releases through other highly probable vents are monitored by the ventilation monitoring system. A radiation monitoring system is also provided on the main stack which monitors all ventilation air releases from the plant. The process liquid radiation monitors are not required, because all liquid wastes are monitored by batch sampling before a controlled release. Limits are expressed on the offgas vent system, liquid radwaste system, and solid radwaste system so that the planned releases of radioactive materials comply with the limits given in 10 CFR 20, 10 CFR 50, and 10 CFR 71 (related unacceptable safety result 1-1).

15A.6.2.4 Operational Safety Evaluations

State D

In State D the reactor vessel head is on and the reactor is not shutdown. Applicable planned operations are achieving criticality, heatup, power operation and achieving shutdown (Events 2, 3, 4, and 5, respectively).

Figure 15A.6-6 relates safety actions for planned operations, corresponding plant systems, and events for which the safety actions are necessary. The required safety actions for planned operation in State D are as follows:

Safety Actions

Radioactive material release control

Core coolant flow rate control

Core power level control

Core neutron flux distribution control

Reactor vessel water level control

Reactor vessel pressure control

Nuclear system temperature control

Nuclear system water quality control

Nuclear system leakage control

Core reactivity control

Rod worth control

Drywell pressure and temperature control

Spent fuel storage shielding, cooling, and reactivity control

15A.6.5 Design Basis Accidents

15A.6.5.1 General

The safety requirements and protection sequences for accidents are described in the following paragraphs for Events 40 through 49. The protection sequence block diagrams show the safety actions and the sequence of front-line safety systems used for the accidents (refer to Figures 15A.6-40 through 15A.6-49). The auxiliaries for the front-line safety systems are indicated in the auxiliary diagrams (Figures 15A.6-1 and 15A.6-2) and the commonality of auxiliary diagrams (Figures 15A.6-60 through 15A.6-65).

Event 42 - Loss-of-Coolant Accidents Resulting from Postulated Piping Breaks Within RPCB Inside Containment (DBA-LOCA)

Pipe breaks inside the containment are considered only when the nuclear system is significantly pressurized (States C and D). The result is a release of steam and water into the containment. Consistent with NSOA criteria, the protection requirements consider all size line breaks including larger liquid recirculation loop piping down to small steam instrument line breaks. The most severe cases are the circumferential break of the largest (liquid) recirculation system pipe and the circumferential break of the largest (steam) main steam line.

As shown in Figure 15A.5-42, in operating State C (reactor shut down, but pressurized), a pipe break accident up to the DBA can be accommodated within the nuclear safety operational criteria through the various operations of the main steam line isolation valves, emergency core cooling systems (HPCS, ADS, LPCI, CSCS, and LPCS), containment and reactor vessel isolation control system, reactor/shield/auxiliary buildings, standby gas treatment system, main control room heating, cooling and ventilation system, MSIV-LCS, emergency service water systems, hydrogen control system, equipment cooling systems, and the incident detection circuitry. For small pipe breaks inside the containment, pressure relief is effected by the nuclear system pressure relief system, which transfers decay heat to the suppression pool. For large breaks, depressurization takes place through the break itself. In State D (reactor not shut down, but pressurized), the same equipment is required as in State C but, in addition, the reactor protection system and the control rod drive system must operate to scram the reactor. The limiting items, on which the operation of the above equipment is based, are the allowable fuel cladding temperature and the containment pressure capability. The control rod drive housing supports are considered necessary whenever the system is pressurized to prevent excessive control rod movement through the bottom of the reactor pressure vessel following the postulated rupture of one control rod drive housing (a lesser case of the design basis loss-of coolant accident and a related preventive of a postulated rod ejection accident).

After completion of the automatic action of the above equipment, manual operation of the RHRS (suppression pool cooling mode) and ADS or relief valves (controlled depressurization) is required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling. (*these are control room actions, if manual isolation is required an operator would need to go to the auxiliary building MCC cabinets to open containment isolation breakers*)

Events 43, 44, 45 - Loss of Coolant Accidents (LOCA) Resulting from Postulated Pipe Breaks - Outside Containment

Pipe break accidents outside the containment are assumed to occur any time the nuclear system is pressurized (States C and D). This accident is most severe during operation at high power (State D). In State C, this accident becomes a subset of the State D sequence.

The protection sequences for the various possible pipe breaks outside the containment are shown in Figures 15A.6-43, 15A.6-44, 15A.6-45. The sequences also show that for small breaks (breaks not requiring immediate action) the reactor operator can use a large number of process indications to identify the break and isolate it.

In operating State D (reactor not shut down, but pressurized), scram is accomplished through operation of the reactor protection system and the control rod drive system. Reactor vessel isolation is accomplished through operation of the main steam line isolation valves and the containment and reactor vessel isolation control system.

For a main steam line break, initial core cooling is accomplished by either the HPCS or the automatic depressurization system (ADS) or manual relief valve operation in conjunction with the LPCS, or LPCI. These systems provide parallel paths to effect initial core cooling, thereby satisfying the single-failure criterion. Extended core cooling is accomplished by the single failure proof, parallel combination of LPCS, HPCS, and LPCI systems. **The ADS or relief valve system operation and the RHRS suppression pool cooling mode (both manually operated) are required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling. (*Control Room Actions*)**

Event 46 - Gaseous Radwaste System Leak or Failure

It is assumed that the line leading to the steam jet air ejector fails near the main condenser. This results in activity normally processed by the offgas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment. This failure results in a loss-of-flow signal to the offgas system. This event can be considered only under States C and D, and is shown in Figure 15A.6-46.

The reactor operator initiates a normal shutdown of the reactor to reduce the gaseous activity being discharged. A loss of main condenser vacuum will result (timing depending on leak rate) in a main turbine trip and ultimately a reactor shutdown. Refer to Event 26 for reactor protection sequence (see Figure 15A.6-26).

Event 47 - Augmented Offgas Treatment System Failure

An evaluation of those events which could cause a gross failure in the offgas system has resulted in the identification of a postulated seismic event, more severe than the one for which the system is designed, as the only conceivable event which could cause significant damage.

The detected gross failure of this system will result in manual isolation of this system from the main condenser. The isolation results in high main condenser pressure and ultimately a reactor scram. Protective sequences for the event are shown in Figure 15A.6-46.

The undetected postulated failure soon results in a system isolation necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in Event 26 (see Figure 15A.6-26).

Event 48 - Liquid Radwaste System Leak or Failure

Releases which could occur inside and outside of the containment, not covered by Events 40, 41, 42, 43, 44, 45, 47, and 48 will probably include small spills and equipment leaks of radioactive materials inside structures housing the subject process equipment. Conservative values for leakage have been assumed and evaluated in the plant under routine releases. The offsite dose that results from any small spill which could occur outside containment will be negligible in comparison to the dose resulting from the accountable (expected) plant leakage.

The protective sequences for this event are provided in Figure 15A.6-48.

Event 49 - Liquid Radwaste System - Storage Tank Failure

An unspecified event causes the complete release of the average radioactivity inventory in the storage tank containing the largest quantities of significant radionuclides from the liquid radwaste system. This is assumed to be one of the concentrate waste tanks in the radwaste building. The airborne radioactivity released during the accident is assumed to bypass directly to the environment via the main plant vent.

The postulated events that could cause release of the radioactive inventory of the concentrate waste tank include cracks in the vessels and an operator error. The possibility of small cracks and consequent low-level release rates receives primary consideration in system and component design. The concentrate waste tank is designed to operate at atmospheric pressure and 200° F maximum temperature so the possibility of failure is considered small. A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instruction. The manually operated drain valve in series with the tank discharge valve prevents inadvertent tank draining. Should a release of liquid radioactive wastes occur, floor drain sump pumps in the floor of the radwaste building will receive a high water level alarm, activate automatically, and remove the spilled liquid to a contained storage tank.

TABLE 15A.6-4 DESIGN BASIS ACCIDENTS

NSOA Event No.	Event Description
40	Control Rod Drop Accident
41	Fuel Handling Accident
42	Loss-of-Coolant Accident Resulting From Spectrum of Postulated Piping Breaks Within the RPCE Inside Containment
43	Small, Large, Steam and Liquid Piping Breaks Outside Containment
44	Instrument Line Break Outside Drywell
45	Feedwater Line Break Outside Containment
46	Gaseous Radwaste System Leak or Failure
47	Augmented Off-Gas Treatment System Failure
48	Liquid Radwaste System Leak or Failure
49	Liquid Radwaste System Storage Tank Failure