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June 27, 2001

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Technical Specification Change Request
TSCR-037 - Alternative Source Term

References: 1.) NG-00-1504, Technical Specification Change Request (TSCR-037),
Alternative Source Term, dated October 19, 2000
2.) Amendment 237 issued April 16, 2001, Regarding Secondary
Containment Operability during Movement of Irradiated Fuel and
Core Alterations

File: A-117, A-225

Reference 1 submitted an amendment request for the Duane Arnold Energy Center (DAEC) to adopt the Alternate Source Term (AST) as defined in NUREG-1465. That request included "marked-up" TS pages.

To expedite the Staff's review, Nuclear Management Company, LLC (NMC) requested review and approval of the portion of that request regarding the Fuel Handling Accident and its attendant changes to the DAEC Technical Specifications (TS) regarding the movement of irradiated fuel assemblies in secondary containment. Reference 2 approved that portion of the application and issued those TS pages.

The attachment to this letter contains "clean typed" pages for the remaining TSCR-037 pages which have not yet been issued by the NRC.

A copy of this submittal is being forwarded to our appointed state official pursuant to 10 CFR Section 50.91.

Should you have any questions regarding this matter, please contact this office.

This letter is true and accurate to the best of my knowledge and belief.


NUCLEAR MANAGEMENT COMPANY, LLC

By Keith D. Young for
Gary Van Middlesworth
DAEC Site Vice-President

State of Iowa
(County) of Linn

Signed and sworn to before me on this 27th day of June, 2001,

by Keith D. Young.

Nancy S. Franck
Notary Public in and for the State of Iowa

Commission Expires

Attachment

cc: R. Anderson (NMC) (w/o)
B. Mozafari (NRC-NRR) (w/a)
J. Dyer (Region III) (w/a)
D. McGhee (State of Iowa) (w/a)
NRC Resident Office (w/a)
Docu (w/a)

1.1 Definitions (continued)

CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/ml), that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC RPT) SYSTEM RESPONSE TIME	The EOC RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint to actuation of the breaker secondary (auxiliary) contact. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

SURVEILLANCE REQUIREMENTS

-----NOTE-----

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the other channel is OPERABLE.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 5 mR/hr.	24 months
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity $\leq 0.2 \mu\text{Ci/gm}$.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Reactor coolant specific activity $> 0.2 \mu\text{Ci/gm}$ and $\leq 2.0 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.</p>	<p>-----NOTE----- LCO 3.0.4 is not applicable. -----</p> <p>A.1 Determine DOSE EQUIVALENT I-131.</p> <p><u>AND</u></p> <p>A.2 Restore DOSE EQUIVALENT I-131 to within limits.</p>	<p>Once per 4 hours</p> <p>48 hours</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p> <p><u>OR</u></p> <p>Reactor Coolant specific activity $> 2.0 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.</p>	<p>B.1 Determine DOSE EQUIVALENT I-131.</p> <p><u>AND</u></p> <p>B.2.1 Isolate all main steam lines.</p> <p><u>OR</u></p>	<p>Once per 4 hours</p> <p>12 hours</p> <p>(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	B.2.2.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.6.1 -----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity is $\leq 0.2 \mu\text{Ci/gm}$.	7 days

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation in normal operating conditions, the reactor coolant acquires radioactive materials due to release of fission products from "tramp" uranium on the outside of the fuel cladding and activation of corrosion products in the reactor coolant. There is the potential for fuel leaks into the reactor coolant that will contribute to the level of radioactive materials due to release of fission products. These radioactive materials in the reactor coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 50.67 (Ref. 1).

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR 50.67 limit.

APPLICABLE
SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the UFSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a Main Steam Line Break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

primary coolant ensure that the 2 hour Total Effective Dose Equivalent (TEDE) doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR 50.67.

The limit on specific activity is plant specific analysis of the radiological consequences of a Main Steam Line Break DBA.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to $\leq 0.2 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 50.67 limits.

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is $\leq 2.0 \mu\text{Ci/gm}$, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 $\mu\text{Ci/gm}$ within 48 hours, or if at any time it is > 2.0 $\mu\text{Ci/gm}$, it must be determined at least once every 4 hours and all the main steam lines, including the main steam line drains, must be isolated within 12 hours. Isolating the main steam lines including drains, precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 50.67 during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without

(continued)

BASES

ACTIONS

B.1, B.2.1, B.2.2.1, and B.2.2.2 (continued)

challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The analysis is performed using filtrate from a 0.45 μ filter. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR 50.67
 2. UFSAR, Sections 15.6.5 and 15.10.3.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SBGT) System

BASES

BACKGROUND

The SBGT System is required by the UFSAR (Ref. 1). The function of the SBGT System is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The SBGT System consists of two fully redundant subsystems, each with its own set of ductwork, dampers, charcoal filter train, and controls.

Each charcoal filter train consists of (components listed in order of the direction of the air flow):

- a. A demister or moisture separator;
- b. An electric heater;
- c. A prefilter;
- d. A High Efficiency Particulate Air (HEPA) filter;
- e. A charcoal adsorber;
- f. A second HEPA filter; and
- g. A centrifugal fan.

HEPA filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place testing of the HEPA filters and charcoal adsorbers is performed under the DAEC Ventilation Filter Testing Program (ITS 5.5.7). If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 50.67 guidelines for the accidents analyzed, as the UFSAR Section 15.6.6 for the loss-of-coolant accident shows compliance with

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BASES

BACKGROUND
(continued)

10 CFR 50.67 guidelines with an assumed efficiency of 99% for the adsorber. Operation of the fans significantly different from the design flow envelope will change the removal efficiency of the HEPA filters and charcoal adsorbers.

The sizing of the SBGT System equipment and components is based on the results of an infiltration analysis, as well as an exfiltration analysis of the secondary containment. The internal pressure of the SBGT System boundary region is maintained at a negative pressure of at least 0.25 inches water gauge (as determined by averaging pressure readings from different faces of the Secondary Containment boundary) when the system is in operation, which represents an internal pressure that ensures zero exfiltration of air from the building when exposed to calm wind conditions (< 15 mph). Maintaining a negative pressure of 0.25 inches water gauge under calm wind conditions ensures a negative pressure under worst case conditions. Therefore, a negative pressure of 0.25 inches water gauge includes some margin to a negative pressure that ensures zero exfiltration.

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SBGT System automatically starts and operates in response to secondary containment isolation actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, both charcoal filter train fans start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.

APPLICABLE
SAFETY ANALYSES

The design basis for the SBGT System is to mitigate the consequences of a loss of coolant accident (Ref. 3). For all events analyzed, the SBGT subsystem is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material

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B 3.7 PLANT SYSTEMS

B 3.7.4 Standby Filter Unit (SFU) System

BASES

BACKGROUND

The SFU System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of the SFU System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of outside supply air. Each subsystem consists of a demister, an electric heater, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a fan, and the associated ductwork and dampers. Demisters remove water droplets from the airstream. HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

The SFU System is a standby system, parts of which also operate during normal unit operations to maintain the control room environment. Upon receipt of the initiation signal (indicative of conditions that could result in radiation exposure to control room personnel), the SFU System automatically starts and a system of dampers isolates the control building to prevent infiltration of contaminated air into the control room. Outside air is taken in at the normal ventilation intake and is passed through one of the charcoal adsorber filter subsystems for removal of airborne radioactive particles before being mixed with the recirculated air. The air (outside and/or recirculated) is cooled by Air Conditioning (AC) units supplied by the Control Building Chillers (CBCs). The SFUs and AC units share common ductwork such that either SFU may supply outside air to either AC unit. However, the CBCs and AC units are addressed as part of LCO 3.7.5, "Control Building Chiller System."

The SFU System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem total effective dose equivalent (TEDE). A single SFU subsystem will

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B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensable gases are collected in the condenser, then exhausted through the Steam Jet Air Ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the 30 minute holdup line.

APPLICABLE
SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Steam-Line Break Accident (Roof Top Release). This analysis assumes the concentrations of radionuclides in the reactor water are those associated with an assumed stack gas release limit of 1,000,000 $\mu\text{Ci}/\text{sec}$ (1.0 Ci/sec)(Ref. 3).

The main condenser offgas limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Restricting the gross radioactivity rate of noble gases from the main condenser (Ref. 1) provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part 50.67 (Ref. 2) in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50. The offgas

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BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.6.1

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by $\geq 50\%$ after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable, based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.

REFERENCES

1. UFSAR, Section 11.3.3.
 2. 10 CFR 50.67.
 3. UFSAR, Section 15.10.3.
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B 3.7 PLANT SYSTEMS

B 3.7.8 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the UFSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 15.7.1 (Ref. 2).

APPLICABLE SAFETY ANALYSES The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are well below the guideline limits of 10 CFR 50.67 (Ref. 3) and meet the exposure guidelines of Regulatory Guide 1.183 (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in UFSAR, Section 15.7.1.4 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the UFSAR, Section 9.1.2.3.3.4 (Ref. 6). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

REFERENCES

1. UFSAR, Section 9.1.2.
 2. UFSAR, Section 15.7.1.
 3. 10 CFR 50.67.
 4. Regulatory Guide 1.183.
 5. UFSAR, Section 15.7.1.4.
 6. UFSAR, Section 9.1.2.3.3.4.
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B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Ref. 1). Sufficient iodine activity would be retained to limit offsite doses from the accident to less than Regulatory Guide 1.183 limits.

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment. A minimum water level of 23 ft allows a decontamination factor of 100 to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory.

Analysis of the fuel handling accident inside containment is described in Reference 1. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained within allowable limits (Ref. 2).

RPV water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV, ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. This verification can be performed using the normal RPV water level indication, which is referenced to Top of Active Fuel (TAF). Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 1).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. UFSAR, Section 15.7.1.
 2. Regulatory Guide 1.183.
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