

rules, regulations, and orders of the Commission, now or hereafter applicable; and is subject to the additional conditions specified and incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady-state reactor core power levels not in excess of 2700 megawatts-thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 281, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

- (a) For Surveillance Requirements (SRs) that are new, in Amendment 227 to Facility Operating License No. DPR-53, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 227. For SRs that existed prior to Amendment 227, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 227.

(3) Additional Conditions

The Additional Conditions contained in Appendix C as revised through Amendment No. 267 are hereby incorporated into this license. Calvert Cliffs Nuclear Power Plant, Inc. shall operate the facility in accordance with the Additional Conditions.

(4) Secondary Water Chemistry Monitoring Program

The Calvert Cliffs Nuclear Power Plant, Inc., shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

- a. Identification of a sampling schedule for the critical parameters and control points for these parameters;
- b. Identification of the procedures used to quantify parameters that are critical to control points;

C. This license is deemed to contain and is subject to the conditions set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and the rules, regulations, and orders of the Commission, now and hereafter applicable; and is subject to the additional conditions specified and incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor steady-state core power levels not in excess of 2700 megawatts-thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 258, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(a) For Surveillance Requirements (SRs) that are new, in Amendment 201 to Facility Operating License No. DPR-69, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 201. For SRs that existed prior to Amendment 201, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 201.

(3) Less Than Four Pump Operation

The licensee shall not operate the reactor at power levels in excess of five (5) percent of rated thermal power with less than four (4) reactor coolant pumps in operation. This condition shall remain in effect until the licensee has submitted safety analyses for less than four pump operation, and approval for such operation has been granted by the Commission by amendment of this license.

(4) Environmental Monitoring Program

If harmful effects or evidence of irreversible damage are detected by the biological monitoring program, hydrological monitoring program, and the radiological monitoring program specified in the Appendix B Technical Specifications, the licensee will provide to the staff a detailed analysis of the problem and a program of remedial action to be taken to eliminate or significantly reduce the detrimental effects or damage.

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include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

Analog Channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY, including alarm and/or trip functions.

Bistable Channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

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 parameter 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/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The TEDE (Total Effective Dose Equivalent) inhalation dose conversion factors used for this calculation shall be those listed in Table 2.1 in the column headed "effective" of Federal Guidance Report 11, ORNL, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

\bar{E} -AVERAGE DISINTEGRATION ENERGY

\bar{E} shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the

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sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. 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. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

L_a

The maximum allowable containment leakage rate, L_a , shall be 0.16% of containment air weight per day at the calculated peak containment pressure (P_a).

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically

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located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolts specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE-OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s):

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Specific Activity

LCO 3.4.15 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) \geq 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
<p>A. DOSE EQUIVALENT I-131 > 0.5 μCi/gm.</p>	<p>----- NOTE ----- LCO 3.0.4.c is applicable. -----</p>		
	<p>A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.15-1.</p>		<p>Once per 4 hours.</p>
	<p><u>AND</u> A.2 Restore DOSE EQUIVALENT I-131 to within limit.</p>		<p>100 hours</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.15.2 -----NOTE----- Only required to be performed in MODE 1. -----</p> <p>Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq 0.5 \mu\text{Ci/gm}$.</p>	<p>14 days</p> <p><u>AND</u></p> <p>Between 2 and 6 hours after THERMAL POWER change of $\geq 15\%$ RTP within a 1 hour period</p>
<p>SR 3.4.15.3 -----NOTE----- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. -----</p> <p>Determine \bar{E} from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.</p>	<p>184 days</p>

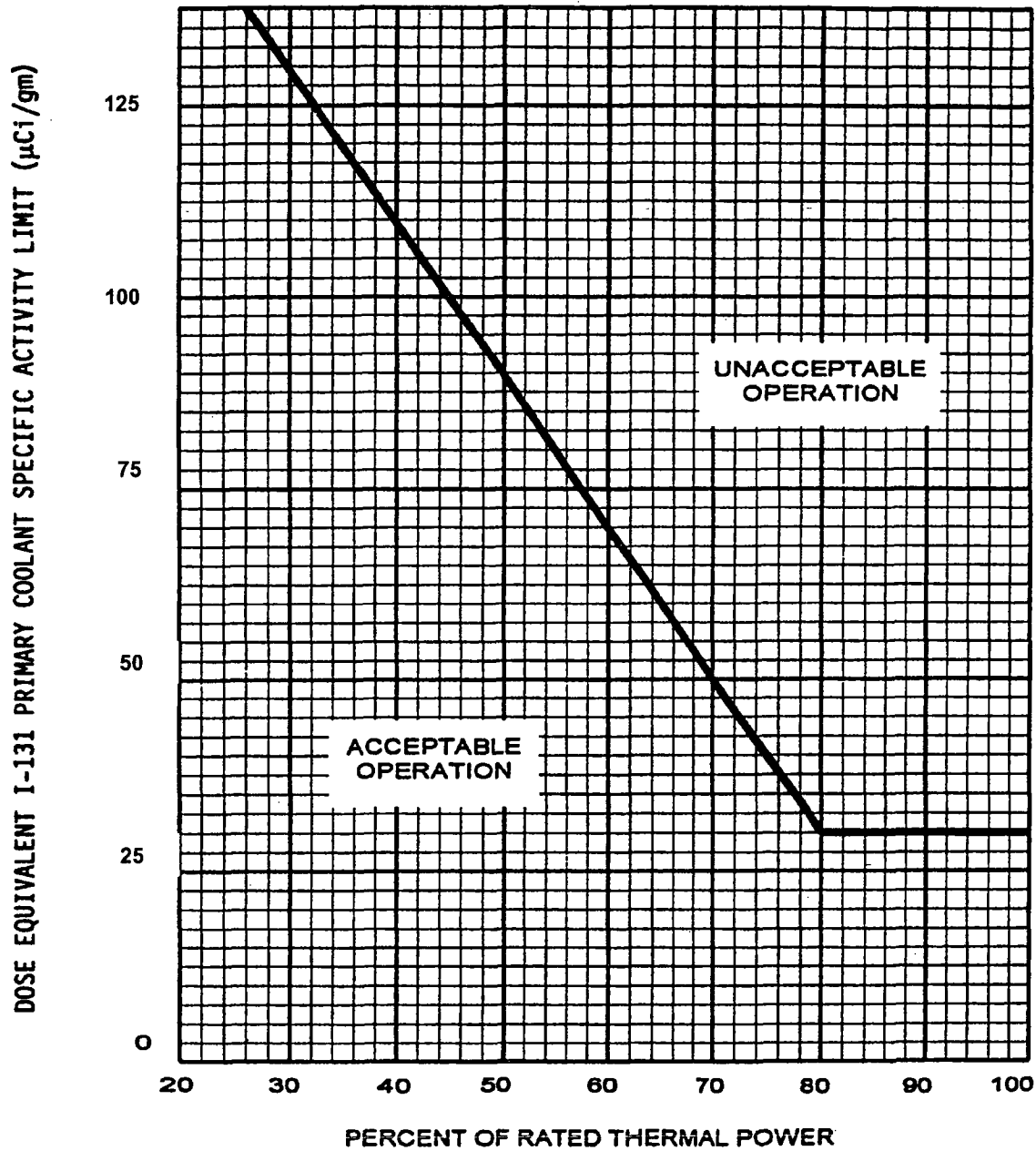


Figure 3.4.15-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit
Versus Percent of RATED THERMAL POWER With Reactor Coolant
Specific Activity $\geq 0.5 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.11.1 Verify an OPERABLE SFPEVS train is in operation.	12 hours
SR 3.7.11.2 Deleted	
SR 3.7.11.3 Verify each SFPEVS fan can maintain a measurable negative pressure with respect to adjacent areas.	24 months

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LC0 3.9.3 The containment penetrations shall be in the following status:

- a. 1. The equipment hatch closed and held in place by a minimum of four bolts, or
- 2. The containment outage door is capable of being closed under administrative control;
- b. One door in the emergency air lock is closed;

----- NOTE -----
The emergency air lock temporary closure device can be used in place of an emergency air lock door.

- c. The personnel air lock shall be either:
 - 1. closed by one personnel air lock door, or
 - 2. capable of being closed by an OPERABLE personnel air lock door under administrative control.
- d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:
 - 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
 - 2. capable of being closed by an OPERABLE Containment Purge Valve Isolation System.

----- NOTE -----
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.3.1 Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2 Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

5.5 Programs and Manuals

- a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass $\leq 1.0\%$ ($\leq 0.05\%$ for the CREVS only) when tested in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975, at the system flowrate specified as follows $\pm 10\%$:

<u>ESF Ventilation System</u>	<u>Flowrate</u>
Control Room Emergency Ventilation System (CREVS)	10,000 cfm
Penetration Room Exhaust Ventilation System (PREVS)	2,000 cfm
IRS	20,000 cfm

- b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass $\leq 1.0\%$ when tested in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975, at the system flowrate specified as follows $\pm 10\%$:

<u>ESF Ventilation System</u>	<u>Flowrate</u>
CREVS	10,000 cfm
PREVS	2,000 cfm
IRS	20,000 cfm

- c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and greater than or equal to the relative humidity specified as follows:

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<u>ESF Ventilation System</u>	<u>Penetrations</u>	<u>RH</u>
CREVS	4.5%	70%
PREVS	34.5%	95%
IRS	34.5%	95%

- d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate specified as follows $\pm 10\%$:

<u>ESF Ventilation System</u>	<u>Delta P</u>	<u>Flowrate</u>
CREVS	6 inwg	10,000 cfm
PREVS	6 inwg	2,000 cfm
IRS	6 inwg	20,000 cfm

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides control for potentially explosive gas mixtures contained in the Waste Gas Holdup System and the quantity of radioactivity contained in gas storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in the ODCM.

The program shall include:

- a. The limits for concentrations of oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and
- b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than or equal to 58,500 curies noble gases (considered as Xe-133).

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The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance Frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A Diesel Fuel Oil Testing Program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
 1. An American Petroleum Institute gravity or an absolute specific gravity within limits,
 2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
 3. Water and sediment $\leq 0.05\%$.
- b. Within 31 days following addition of new fuel oil to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil; and
- c. Total particulate concentration of the fuel oil, when determined by gravimetric analysis based on ASTM D2276-1989, is ≤ 10 mg/l when tested every 92 days.
- d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Frequencies.

5.5 Programs and Manuals

5.5.14 Technical Specifications Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the Technical Specifications shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
 1. A change in the Technical Specifications incorporated in the license; or
 2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.
- d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into Limiting Condition for Operation (LCO) 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and

5.5 Programs and Manuals

Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

- a. Provisions for cross-train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
- b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;
- c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
- d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

- a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or
- b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, including errata, as modified by the following exceptions:

- a. Nuclear Energy Institute (NEI) 94-01 – 1995, Section 9.2.3: The first Unit 1 Type A test performed after the June 15, 1992 Type A test shall be performed no later than June 14, 2007.
- b. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.
- c. Unit 2 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a , is 49.4 psig. The containment design pressure is 50 psig.

The maximum allowable containment leakage rate, L_a , shall be 0.16 percent of containment air weight per day at P_a .

Leakage rate acceptance criteria are:

- a. Containment leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first unit startup following testing, in accordance with this program, the leakage rate acceptance criterion are $\leq 0.60 L_a$ for Types B and C tests and $\leq 0.75 L_a$ for Type A tests.