TABLE 7—THRESHOLD VALUES FOR THE OUTLIER DEVIATION TEST (SIGNIFICANCE LEVEL = 1%)

Number of available data points (n)	Second largest allowable normalized residual value (r*)	Largest allowable normalized residual value (r*)
3	1.55	2.71
4	1.73	2.81
5	1.84	2.88
6	1.93	2.93
7	2.00	2.98
8	2.05	3.02
9	2.11	3.06
10	2.16	3.09
11	2.19	3.12
12	2.23	3.14
13	2.26	3.17
14	2.29	3.19
15	2.32	3.21

[75 FR 23, Jan. 4, 2010, as amended at 75 FR 5495, Feb. 3, 2010; 75 FR 10411, Mar. 8, 2010; 75 FR 72653, Nov. 26, 2010]

§ 50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

(a) *Applicability.* The requirements of this section apply to all commercial light-water-cooled nuclear power plants, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted.

(b) *Definition.* For purposes of this section, *Anticipated Transient Without Scram* (ATWS) means an anticipated operational occurrence as defined in appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of appendix A of this part.

(c) *Requirements.* (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

(2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).

(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

(4) Each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must

be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

(5) Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

(6) Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Commission as specified in § 50.4.

(d) *Implementation.* For each light-water-cooled nuclear power plant operating license issued before September 27, 2007, by 180 days after the issuance of the QA guidance for non-safety related components, each licensee shall develop and submit to the Commission, as specified in § 50.4, a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each shall include an explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outage after July 26, 1984, or the date of issuance of a license authorizing operation above 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee. For each light-water-cooled nuclear power plant operating license application submitted after September 27, 2007, the applicant shall submit information in its final safety analysis report demonstrating how it will comply with paragraphs (c)(1) through (c)(5) of this section.

[49 FR 26044, June 26, 1984; 49 FR 27736, July 6, 1984, as amended at 51 FR 40310, Nov. 6, 1986; 54 FR 13362, Apr. 3, 1989; 61 FR 39301, July 29, 1996; 72 FR 49500, Aug. 28, 2007]

§ 50.63 Loss of all alternating current power.

(a) *Requirements.* (1) Each light-water-cooled nuclear power plant licensed to operate under this part, each light-water-cooled nuclear power plant licensed under subpart C of 10 CFR part 52 after the Commission makes the finding under § 52.103(g) of this chapter, and each design for a light-water-cooled nuclear power plant approved under a standard design approval, standard design certification, and manufacturing license under part 52 of this chapter must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2. The specified station blackout duration shall be based on the following factors:

(i) The redundancy of the onsite emergency ac power sources;

(ii) The reliability of the onsite emergency ac power sources;

(iii) The expected frequency of loss of offsite power; and

(iv) The probable time needed to restore offsite power.

(2) The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Licensees are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

(b) *Limitation of scope.* Paragraph (c) of this section does not apply to those plants licensed to operate prior to *July 21, 1988,* if the capability to withstand station blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC.

(c) *Implementation*—(1) *Information Submittal.* For each light-water-cooled nuclear power plant operating license application submitted after September 27, 2007, the applicant shall submit the information defined below in its final safety analysis report.

(i) A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section;