United States Nuclear Regulatory Commission
Attachment II to Serial: RNP-RA/99-0198
27 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

INSTRUCTIONS FOR REMOVAL AND
INSERTION OF PAGES TO THE LISTS OF EFFECTIVE
PAGES, OPERATING LICENSE, TECHNICAL SPECIFICATIONS, AND BASES

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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

INSTRUCTIONS FOR REMOVAL AND
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3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20. Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified below:

   A. Maximum Power Level

   The licensee is authorized to operate the facility at a steady state reactor core power level not in excess of 2300 megawatts thermal.

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 185 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   (1) For Surveillance Requirements (SRs) that are new in Amendment 176 to Final Operating License DPR-23, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 176. For SRs that existed prior to Amendment 176, 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 176.

   C. Reports

   Carolina Power & Light Company shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

   Carolina Power & Light Company shall keep facility operating records in accordance with the requirements of the Technical Specifications.

4. Additional Conditions

   The Additional Conditions contained in Appendix B, as revised through Amendment No. 185, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the additional conditions.

5. This license is effective as of the date of issuance and shall expire at midnight July 31, 2010.

Attachment
Appendix A - Technical Specifications

Date of Issuance: JUL 31 1970

3.7 PLANT SYSTEMS

3.7.8 Ultimate Heat Sink (UHS)

LCO 3.7.8 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

------------------------------------NOTES------------------------------------

1. Conditions A and B and associated Required Actions and Completion Times shall only be applicable prior to, and on September 30, 1999.

2. Condition C and associated Required Actions and Completion Times shall only be applicable after September 30, 1999.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Service water temperature > 95°F.	A.1 Restore service water temperature to ≤ 95°F.	72 hours
	<u>AND</u>	
	A.2 Verify service water temperature is ≤ 99°F.	1 hour <u>AND</u> Once per hour thereafter

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, Richland WA 99352.

19. EMF-92-081(A), latest Revision and Supplements, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," Siemens Power Corporation - Nuclear Division, Richland, WA 99352.

20. EMF-92-153(P)(A), Revision 0 and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, Richland WA 99352, March 7, 1994.

21. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986.

22. EMF-96-029(P)(A), Volume 1, Volume 2, and Attachment, "Reactor Analysis System for PWRs, "Siemens Power Corporation, Richland WA, 99352, January 1997.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or H of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

(continued)

5.6 Reporting Requirements (continued)

5.6.7 Tendon Surveillance Report

a. Notification of a pending sample tendon test, along with detailed acceptance criteria, shall be submitted to the NRC at least two months prior to the actual test.

b. A report containing the sample tendon test evaluation shall be submitted to the NRC within six months of conducting the test.

5.6.8 Steam Generator Tube Inspection Report

a. A report of the number of tubes plugged in each steam generator shall be submitted to the NRC within 14 days after completion of the tube plugging.

b. A report of the results of the steam generator tube inspection shall be included in the Monthly Operating Report for the period beginning after the final inspection is completed.

Reports shall include:

1. Number and extent of tubes inspected
2. Location and percent of wall thickness penetration for each eddy current indication and any leaks.
3. Identification of tubes plugged.

c. A report of examination results falling in Category C-3 of Table 5.5-1 shall be submitted to the NRC within 30 days, and prior to resumption of plant operation.

A report of investigations conducted to determine cause(s) of the tube degradation and corrective measures taken to prevent recurrence shall be submitted within 90 days following completion of the startup test program.

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B 3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by unit specific documents (Ref. 5) addressing the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category 1 variables.

Type A variables are included in this LCO because they provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for DBAs.

Category 1 variables are the key variables deemed risk significant because they are needed to:

(continued)

BASES

BACKGROUND (continued)

- Determine whether other systems important to safety are performing their intended functions;
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public, and to estimate the magnitude of any impending threat.

These key variables are identified by the HBRSEP Regulatory Guide 1.97 analyses (Ref. 5). These analyses identify the unit specific Type A and Category 1 variables and provide justification for deviating from the NRC proposed list of Category 1 variables.

This LCO also includes certain parameters associated with risk-significant scenarios or mitigating systems as modeled in the HBRSEP Probabilistic Safety Assessment (PSA). These instruments include Auxiliary Feedwater Flow, PORV Position (primary indication), PORV Block Valve Position (primary indication), and Pressurizer Safety Valve Position (primary indication).

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the operability of Regulatory Guide 1.97 Type A and Category 1 variables so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures (these variables are restricted to preplanned actions for the primary success path of DBAs), e.g., loss of coolant accident (LOCA);
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

- Determine whether systems important to safety are performing their intended functions;
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of the NRC Policy Statement. Category 1, non-Type A, instrumentation must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category 1, non-Type A, variables are important for reducing public risk.

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category 1, non-Type A and selected Category 2 and 3 instruments.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident. This capability is consistent with the recommendations of Reference 1.

LCO 3.3.3 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident. The exception to the single failure criterion is the RCS Cold Leg wide range instrumentation which for RCS Loops “B” and “C” is powered from the same instrument power

(continued)

BASES

LCO (continued)

supply. Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

The exception to the two channel requirement is Containment Isolation Valve (CIV) Position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

Type A and Category 1 variables are required to meet Regulatory Guide 1.97 Category 1 (Ref. 2) design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.3-1.

1, 2. <u>Power Range and Source Range Neutron Flux</u>

Power Range and Source Range Neutron Flux indication is provided to verify reactor shutdown. The two ranges are necessary to cover the full range of flux that may occur post accident.

Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

3, 4. <u>Reactor Coolant System (RCS) Hot and Cold Leg Temperatures</u>

RCS Hot Leg Temperatures are Category 1 variables provided for verification of core cooling and long term surveillance. The RCS Cold Leg Temperatures are

(continued)

BASES

LCO

3, 4. Reactor Coolant System (RCS) Hot and Cold Leg Temperatures (continued)

Category 1 in RCS Loop "B" and "C" and Category 3 in RCS Loop "A."

RCS hot leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin will allow termination of safety injection (SI), if still in progress, or reinitiation of SI if it has been stopped. RCS subcooling margin is also used for unit stabilization and cooldown control. The RCS Cold Leg temperatures provide backup/verification indication to the core exit temperature.

In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the unit conditions necessary to establish natural circulation in the RCS. The RCS Loop "A" Cold Leg temperature instrument is a Category 3 variable and does not meet Regulatory Guide 1.97 design criteria for emergency power. RCS Loops "B" and "C" Cold Leg temperature instruments do not meet Regulatory Guide 1.97 design criteria for power redundancy.

5. Reactor Coolant System Pressure (Wide Range)

RCS wide range pressure from the Inadequate Core Cooling Monitor is a Category 1 variable provided for verification of core cooling and RCS integrity long term surveillance.

RCS pressure is used to verify delivery of SI flow to RCS from at least one train when the RCS pressure is below the pump shutoff head. RCS pressure is also used to verify closure of manually closed spray line valves and pressurizer power operated relief valves (PORVs).

In addition to these verifications, RCS pressure is used for determining RCS subcooling margin. RCS subcooling margin will allow termination of SI, if still in progress, or reinitiation of SI if it has been stopped. RCS pressure can also be used:

(continued)

BASES

LCO

5. Reactor Coolant System Pressure (Wide Range) (continued)

- to determine whether to terminate actuated SI or to reinitiate stopped SI;
- to determine when to reset SI and shut off low head SI;
- to manually restart low head SI;
- as reactor coolant pump (RCP) trip criteria; and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.3 (continued)

verify the OPERABILITY of position indication against the actual position of the associated valves.

The Frequency is based upon the known reliability of the Functions and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The affected Functions have no setpoints.

REFERENCES

1. NRC Safety Evaluation Report, H. B. Robinson Steam Electric Plant Unit No. 2, Docket No. 50-261, Conformance to Regulatory Guide 1.97, transmitted to CP&L by letter dated March 5, 1987.

2. Regulatory Guide 1.97, Revision 3, May 1983.

3. NUREG-0737, Supplement 1, "TMI Action Items."

4. CP&L Letter to NRC, "Inadequate Core Cooling Instrumentation, Generic Letter 82-28, NUREG-0737, Item II.F.2, Implementation Letter/License Amendment Request," dated September 16, 1987.

5. CP&L letters dated December 31, 1984, July 18, 1985, July 28, 1985, May 1, 1987, September 9, 1987, and September 14, 1999, regarding the HBRSEP Regulatory Guide 1.97 submittal.

BASES

ACTIONS

B.1 and B.2 (continued)

control power restored. Additionally, Required Action B.2 requires the control power to be removed to the valve within 24 hours. In this condition, the valves could be subject to a spurious single failure that could result in closure of the valve and isolation of an accumulator. During the interval in which control power is restored, the valve remains in its required position. The flow path to FCV-605 may be isolated in lieu of FCV-605 being in the required position. The 24 hour Completion Time is reasonable considering a low probability of a spurious single failure coincident with a LOCA.

C.1 and C.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed

Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of control power or by key locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer codes designed to predict the resultant pressure and temperature transient. The containment pressure analysis indicates the containment peak pressure for the limiting SLB slightly exceeds the peak pressure for the limiting LOCA (Ref. 1).

The initial pressure condition used in the containment analysis was 15 psia (0.3 psig). This resulted in a maximum peak pressure from a LOCA of 40 psig. The containment analysis (Ref. 1) confirms that this calculated peak containment pressure from the limiting LOCA is the same as the P

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The UHS satisfies Criterion 3 of the NRC Policy Statement.
LCO	The UHS is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate for at least 22 days following the design basis LOCA without the loss of NPSH, and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed 95°F and the level should not fall below 218 ft MSL during normal unit operation.
APPLICABILITY	In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES. In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.
ACTIONS	Notes 1 and 2 have been added in the ACTIONS to provide a clear expiration date for Conditions A and B and associated Required Actions and Completion Times, and a date that Condition C and its associated Required Actions and Completion Times will become applicable. Prior to midnight October 1, 1999, if the LCO is not met, refer to Conditions A or B and associated Required Actions and Completion Times. On midnight October 1, 1999, and thereafter, refer only to Condition C if the LCO is not met. <u>A.1</u> When service water temperature is greater than 95°F, it must be restored to ≤ 95°F within 72 hours. This Required Action is necessary to return operation to within the design basis of the Service Water System. The 72 hour Completion Time is acceptable considering the low probability of a Design Basis

(continued)

BASES

BACKGROUND (continued)

The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution System - Operating," and LCO 3.8.10, "Distribution Systems - Shutdown."

Each battery has adequate storage capacity to carry the required load continuously for at least 1 hour following a plant trip and a loss of all AC power (Ref. 2).

There is no sharing between redundant subsystems, such as batteries, battery chargers, or distribution panels.

The battery for Train A DC electrical power subsystem is sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. Battery size is based on 125% of required capacity and, after selection of an available commercial battery, results in a battery capacity in excess of 150% of required capacity. The battery for Train B DC electrical power subsystem is sized to produce required capacity at 91% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. Battery size is based on 110% of required capacity and, after selection of an available commercial battery, results in a battery capacity in excess of 120% of required capacity. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery.

Each Train A and Train B DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from a partial discharge condition to its fully charged state within 24 hours while supplying normal steady state loads discussed in the UFSAR, Chapter 8 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 3), and in the UFSAR, Chapter 15 (Ref. 4), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC

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