UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 4, 2000

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING ENHANCEMENT OF REACTOR COOLANT LEAKAGE DETECTION AND OPERATIONAL LEAKAGE CONSISTENT WITH STANDARD TECHNICAL SPECIFICATIONS (TAC NOS. MA6760 AND MA6761) **(TS 98-10)**

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 259 to Facility Operating License No. DPR-77 and Amendment No. 250 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. These amendments are in response to your application dated June 30, 1999, as supplemented June 16, 2000, and August 3, 2000, which requested approval to update the current Technical Specification (TS) requirements, and appropriate TS Bases sections for reactor coolant system (RCS) leakage detection and RCS operational leakage specifications consistent with the Improved Westinghouse Standard TS (NUREG-1431). The U.S. Nuclear Regulatory Commission staff has reviewed and approved your request.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

William O. Long

for Ronald W. Hernan, Sr. Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures: 1. Amendment No. 259 to License No. DPR-77
2. Amendment No. 250 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

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/RA by Willian O. Long for/
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SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING ENHANCEMENT OF REACTOR COOLANT LEAKAGE DETECTION AND OPERATIONAL LEAKAGE CONSISTENT WITH STANDARD TECHNICAL SPECIFICATIONS (TAC NOS. MA6760 AND MA6761) (**TS 98-10**)

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 259
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 30, 1999, as supplemented on June 16, 2000, and August 3, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 259 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr

for Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: **August 4, 2000**

ATTACHMENT TO LICENSE AMENDMENT NO. 259

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Index Page I	Index Page I
Index Page VI	Index Page VI
Index Page VII	Index Page VII
Index Page XIII	Index Page XIII
1-2	1-2
1-3	1-3
1-7	1-7
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
----	3/4 5-12
----	3/4 5-13

B 3/4 4-4b and c	B 3/4 4-4aa through B 3/4 4-4r
----	B 3/4 5-4
----	B 3/4 5-5
----	B 3/4 5-6
----	B 3/4 5-7

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DEFINITIONS

SECTION PAGE

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

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

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

R145

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3. R207

b. All equipment hatches are closed and sealed.

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3, R180

d. The containment leakage rates are within the limits of Specification 4.6.1.1.c,

e. The sealing mechansim associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE, and

f. Secondary containment bypass leakage is within the limits of Specification 3.6.1.2.

CONTROLLED LEAKAGE

1.8 This definition has been deleted.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position. R205

CORE OPERATING LIMIT REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.14. Unit operation within these operating limits is addressed in individual specifications. R159

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/ gram) which 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 thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURE 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 the methodology for verification have been previously reviewed and approved by NRC.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table I.2.

GASEOUS RADWASTE TREATMENT SYSTEM

1.15 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage, such as that from pump seals or valve packing (except Reactor Coolant Pump Seal Water Injection or Leakoff), that is captured and conducted to collection systems or a sump or collecting tank, or

UNIDENTIFIED LEAKAGE

1.36 UNIDENTIFIED LEAKAGE shall be all leakage (except reactor coolant pump seal water injection or leakoff) that is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.37 An UNRESTRICTED AREA shall be any area, at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commerical, institutional, and/or recreational purposes. |R159 R75

VENTILATION EXHAUST TREATMENT SYSTEM

1.38 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components. |R159

VENTING

1.39 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process. |R159

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection instrumentation shall be OPERABLE:

a. Two lower containment atmosphere radioactivity monitors (gaseous and particulate), and

b. One containment pocket sump level monitor.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS:

a. With both containment pocket sump monitors inoperable, operation may continue for up to 30 days provided SR 4.4.6.2.1 is performed once per 24 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With either or both the gaseous or particulate lower containment atmosphere radioactivity monitors inoperable, operation may continue for up to 30 days provided grab samples of the lower containment atmosphere are analyzed once per 24 hours or SR 4.4.6.2.1 is performed once per 24 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

c. With both containment pocket sump monitors and both lower containment atmosphere radioactivity monitors inoperable, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection instrumentation shall be demonstrated OPERABLE by:

a. Performance of the lower containment atmosphere gaseous and particulate monitor CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Performance of containment pocket sump level monitor CHANNEL CALIBRATION at least once per 18 months.

* Surveillance performance not required until 12 hours after establishment of steady state operation.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day of primary-to-secondary leakage through any one steam generator, and | R226

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be verified to be within each of the above limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours.*

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4. | R16

4.4.6.2.2 Verify steam generator tube integrity is in accordance with the requirements of Technical Specification 3/4.4.5, "Steam Generators."

* Not required to be performed until 12 hours after establishment of steady state operation.

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.3 Leakage from each Reactor Coolant System Pressure Isolation Valve, specified in Table 3.4-1, shall be equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure ≥ 2215 psig and ≤ 2255 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the residual heat removal system flow path when in, or during the transition to or from, the residual heat removal mode of operation.

ACTIONS:

a. With one or more flow paths with leakage from one or more Reactor Coolant System Pressure Isolation Valves greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual, deactivated automatic, or check valve* and restore the inoperable Reactor Coolant System Pressure Isolation Valve to OPERABLE status within the following 68 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Separate entry into the above ACTION is allowed for each flow path.

c. Entry into the applicable ACTIONS for systems made inoperable by an inoperable Reactor Coolant System Pressure Isolation Valve is required.

SURVEILLANCE REQUIREMENTS

4.4.6.3 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing requirements required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit#:

a. At least once per 18 months

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months.

c. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

Not required to be performed in MODES 3 and 4.

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* Each valve used to satisfy ACTION a must have been verified to meet the Surveillance Requirement 4.4.6.3 and be in the reactor coolant pressure boundary.

\# Not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the Residual Heat Removal flow path when in the shutdown cooling mode of operation.

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.6 SEAL INJECTION FLOW

LIMITING CONDITION FOR OPERATION

3.5.6 Reactor coolant pump seal injection flow shall be within the limits of Figure 3.5.6-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

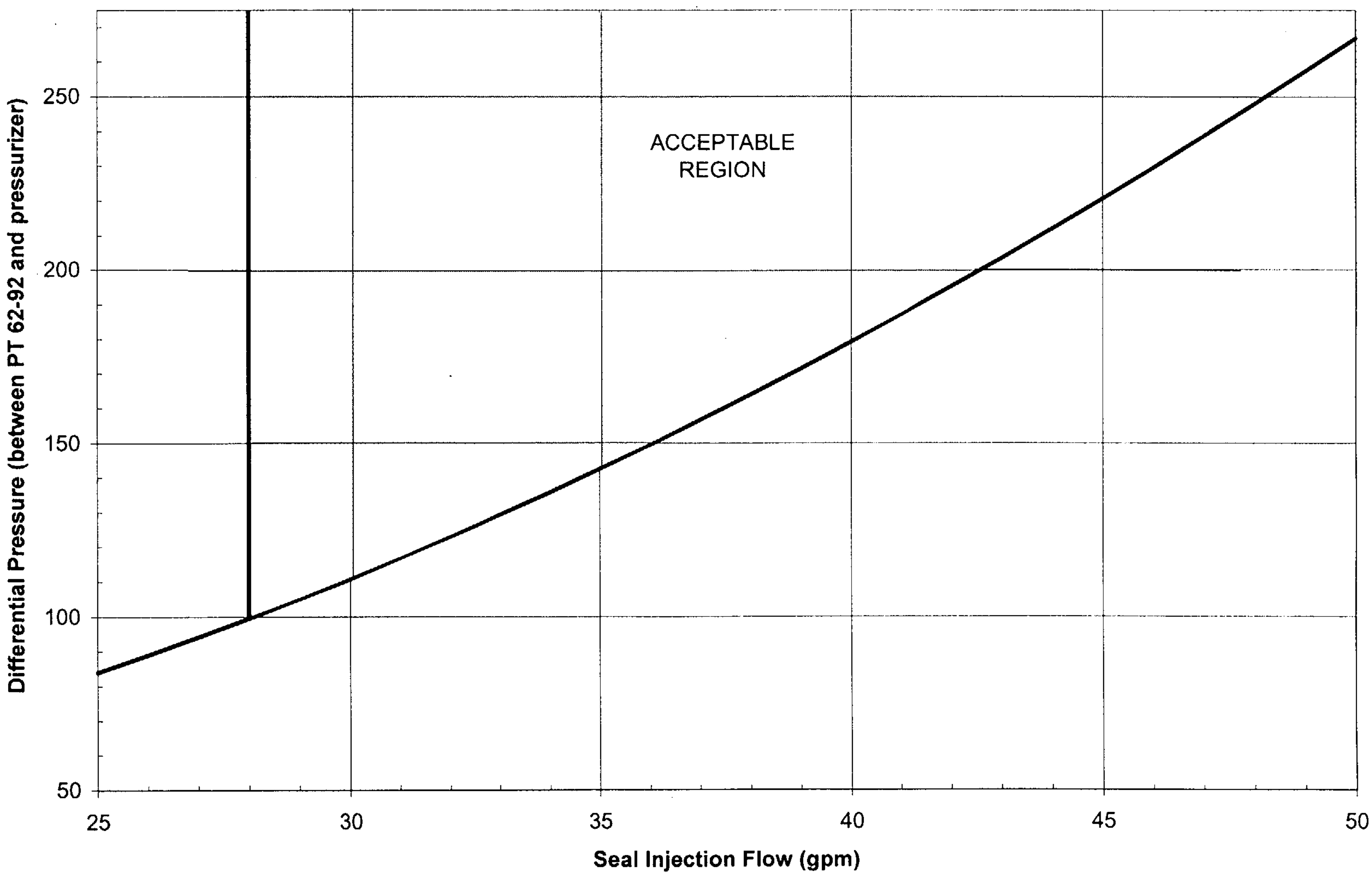
With reactor coolant pump seal injection flow not within limits, adjust manual seal injection throttle valves to give a flow within limit in accordance with Surveillance Requirement 4.5.6 within 4 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.6 At least once per 31 days* verify manual seal injection throttle valves are adjusted to give a flow within the emergency core cooling system safety analysis limits in Figure 3.5.6-1.

*This surveillance is not required to be performed until 4 hours after the reactor coolant system pressure stabilizes at ≥ 2215 psig and ≤ 2255 psig.

FIGURE 3.5.6-1
Seal Injection Flow Limits



3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION INSTRUMENTATION

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of reactor coolant system (RCS) leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all UNIDENTIFIED LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment pocket sump used to collect UNIDENTIFIED LEAKAGE is instrumented to alarm for increases of 1.0 gpm in the normal flow rates within one hour. This sensitivity is acceptable for detecting increases in UNIDENTIFIED LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} mCi/cc radioactivity for particulate monitoring and of 10^{-6} mCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS leakage.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS leakage.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION INSTRUMENTATION

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of reactor coolant system (RCS) leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all UNIDENTIFIED LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment pocket sump used to collect UNIDENTIFIED LEAKAGE is instrumented to alarm for increases of 1.0 gpm in the normal flow rates within one hour. This sensitivity is acceptable for detecting increases in UNIDENTIFIED LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} mCi/cc radioactivity for particulate monitoring and of 10^{-6} mCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS leakage.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS leakage.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer UNIDENTIFIED LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 3). Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS leakage into the containment area is necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public. Exclusions to the requirements of General Design Criteria 4, for the dynamic effects of the RCS piping, have been utilized based on the leak detection capability to identify leaks before a pipe break would occur.

RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS leakage indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, one containment pocket sump monitor, in combination with a gaseous and particulate radioactivity monitor, provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is to be £ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

Action a:

With both containment pocket sump monitors inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, Surveillance 4.4.6.2.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A footnote is added allowing that SR 4.4.6.2.1 is not required to be performed until 12 hours after establishing steady state operation (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup, letdown, and RCP seal injection and return flows). The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required pocket sump monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the frequency and adequacy of the RCS water inventory balance required by Action a.

Action a is modified by a note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

If the requirements of Action a cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 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.

Action b:

With either the gaseous or particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with Surveillance 4.4.6.2.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the containment atmosphere radioactivity monitors.

The 24 hour interval provides periodic information that is adequate to detect leakage. A footnote is added allowing that SR 4.4.6.2.1 is not required to be performed until 12 hours after establishing steady state operation (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup, letdown, and RCP seal injection and return flows). The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

Action b is modified by a note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous and particulate containment atmosphere radioactivity monitor channels are inoperable. This allowance is provided because other instrumentation is available to monitor for RCS leakage.

If the requirements of Action b cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 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.

Action c:

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown to a MODE in which the requirement does not apply is required. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 hours.

SURVEILLANCE REQUIREMENTS

Surveillance 4.4.6.1.a

This surveillance requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence that the monitors are operating properly. The frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

This surveillance requires the performance of a CHANNEL CALIBRATION for the required containment atmosphere radioactivity monitors. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Operating experience has proven that this frequency is acceptable.

This surveillance requires the performance of a CHANNEL FUNCTIONAL TEST on the required containment atmosphere radioactivity monitors. The test ensures that the monitors can perform their functions in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

The surveillance frequencies for these tests are specified in Table 4.3-3.

Surveillance 4.4.6.1.b

This surveillance requires the performance of a CHANNEL CALIBRATION for the required containment pocket sump monitors. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45.
3. FSAR, Sections 5.2.7 "RCBP Leakage Detection Systems" and 12.2.4 "Airborne Radioactivity Monitoring."

3/4.4.6.2 OPERATIONAL LEAKAGE

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational leakage LCO is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified leakage is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary leakage as the initial condition.

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via safety valves for up to 30 minutes. Operator action is taken to isolate the affected steam generator within this time period. The 1 gpm primary to secondary leakage is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm primary to secondary leakage in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction
of these limits). Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 8.21 gpm at atmospheric conditions and 70°F in the faulted loop, which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines. If the projected and cycle distribution of crack indications results in primary-to-secondary leakage greater than 8.21 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 8.21 gpm. R241

The RCS operational leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS operational leakage shall be limited to:

a. <u>PRESSURE BOUNDARY LEAKAGE</u>

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the RCPB. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

b. <u>UNIDENTIFIED LEAKAGE</u>

One gpm of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment pocket

sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the leakage is from the pressure boundary.

c. Primary to Secondary Leakage through Any One Steam Generator (SG)

The 150 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

The 150-gallons per day limit incorporated into Surveillance 4.4.6.2.1 is more restrictive than the standard operating leakage limit and is intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that, should a significant leak be experienced, it will be detected, and the plant shut down in a timely manner. R241

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the RCS Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for reactor coolant PRESSURE BOUNDARY LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

LCO 3/4.4.6.3, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS leakage when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.

ACTIONS

Action a:

If any PRESSURE BOUNDARY LEAKAGE exists, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within the following 30 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

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. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Action b:

UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or primary-to-secondary leakage in excess of the LCO limits must be reduced to within limits within 4 hours. This completion time allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB. If UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or primary to secondary leakage cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within the following 30 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

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. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

Surveillance 4.4.6.2.1

Verifying RCS leakage to be within the LCO limits ensures the integrity of the RCPB is maintained. PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of an RCS water inventory balance. Primary-to-secondary leakage is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup, letdown, and RCP seal injection and return flows). Therefore, a footnote is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Performance of this surveillance within the 12-hour allowance is required to maintain compliance with the provisions of Specification 4.0.3.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment pocket sump level. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3/4.4.6.1, "Leakage Detection Instrumentation."

The 72 hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

Surveillance 4.4.6.2.2

This surveillance provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this surveillance cannot be performed at normal operating conditions.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.

2. Regulatory Guide 1.45, May 1973.

3. FSAR, Section 15.4.3.

3/4.4.6.3 REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE LEAKAGE

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define reactor coolant system (RCS) pressure isolation valves (PIVs) as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment or an unanalyzed accident that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC, "Reactor Safety Study," (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System;

b. Safety Injection System; and

c. Chemical and Volume Control System.

The PIVs are listed in Table 3.4-1.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS PIV leakage is IDENTIFIED LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 6 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

Action a:

The flow path must be isolated. Action a is modified by a note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB.

Action a requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4-hour completion time allows the actions and restricts the operation with leaking isolation valves.

The 72-hour completion time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this action and the low probability of a second valve failing during this period.

If leakage cannot be reduced or the system isolated, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within the following 30 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. 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.

Action b:

Action b provides clarification that each flow path allows separate entry into Action a This is allowed based upon the functional independence of the flow path.

Action c:

Action c requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

SURVEILLANCE REQUIREMENTS

Surveillance 4.4.6.3

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Action a is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month frequency is consistent with 10 CFR 50.55a(g) (Ref. 7) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 6), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the surveillances were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this surveillance. The note that allows this provision is complementary to the frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. 10 CFR 50.2.

2. 10 CFR 50.55a(c).

3. 10 CFR 50, Appendix A, Section V, GDC 55.

4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.

5. NUREG-0677, May 1980.

6. ASME, Boiler and Pressure Vessel Code, Section XI.

7. 10 CFR 50.55a(g).

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with containment concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the containment concentrations to within the Steady State Limits.

3/4.5.6 SEAL INJECTION FLOW

BACKGROUND

The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during safety injection.

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

This LCO ensures that seal injection flow will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of the NRC Policy Statement.

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is established by adjusting the RCP seal injection needle valves to provide a total seal injection flow in the acceptable region of Technical Specification Figure 3.5.6-1. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The flow limits established by Technical Specification Figure 3.5.6-1 are consistent with the accident analysis.

The limits on seal injection flow must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTION

With the seal injection flow exceeding its limit, the amount of charging flow available to the RCS may be reduced. Under this condition, action must be taken to restore the flow to below its limit. The operator has 4 hours from the time the flow is known to be above the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The completion time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is conservative with respect to the completion times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.

When the actions cannot be completed within the required completion time, a controlled shutdown must be initiated. The completion time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown from MODE 3, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

Surveillance 4.5.6

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a flow within the limit ensures that proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. The differential pressure that is above the reference minimum value is established between the charging header (PT 62-92) and the RCS, and total seal injection flow is verified to be within the limits determined in accordance with the ECCS safety analysis (Ref. 3). The seal water injection flow limits are shown in Technical Specification Figure 3.5.6-1. The frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve surveillance frequencies. The frequency has proven to be acceptable through operating experience.

The requirements for charging flow vary widely according to plant status and configuration. When charging flow is adjusted, the positions of the air-operated valves, which control charging flow, are

adjusted to balance the flows through the charging header and through the seal injection header to ensure that the seal injection flow to the RCPs is maintained between 8 and 13 gpm per pump. The reference minimum differential pressure across the seal injection needle valves ensures that regardless of the varied settings of the charging flow control valves that are required to support optimum charging flow, a reference test condition can be established to ensure that flows across the needle valves are within the safety analysis. The values in the safety analysis for this reference set of conditions are calculated based on conditions during power operation and they are correlated to the minimum ECCS flow to be maintained under the most limiting accident conditions.

As noted, the surveillance is not required to be performed until 4 hours after the RCS pressure has stabilized within a ± 20 psig range of normal operating pressure. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the surveillance is timely. Performance of this surveillance within the 4-hour allowance is required to maintain compliance with the provisions of Specification 4.0.3.

REFERENCES

1. FSAR, Chapter 6.3 "Emergency Core Cooling System" and Chapter 15.0 "Accident Analysis".
2. 10 CFR 50.46.
3. Westinghouse Electric Company Calculation CN-FSE-99-48

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 250
License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 30, 1999, as supplemented on June 16, 2000, and August 3, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 250 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

for Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: **August 4, 2000**

ATTACHMENT TO LICENSE AMENDMENT NO. 250

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
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----	3/4 5-13
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----	B 3/4 4-4a
----	B 3/4 4-4b
----	B 3/4 4-4c
----	B 3/4 4-4d
----	B 3/4 4-4e
----	B 3/4 4-4f
----	B 3/4 4-4g
----	B 3/4 4-4h
----	B 3/4 4-4i
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

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

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

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions. R132

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:
   1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or
   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3. R193

b. All equipment hatches are closed and sealed.
c. Each air lock is in compliance with the requirements of Specification 3.6.1.3, R117
d. The containment leakage rates are within the limits of Specification 4.6.1.1.c, and R167
e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE, and
d. Secondary containment bypass leakage is within the limits of Specification 3.6.1.2.

CONTROLLED LEAKAGE

1.8 This definition has been deleted.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position. R191

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.14. Unit operation within these operating limits is addressed in individual specifications. R146

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be: R146

a. Leakage, such as that from pump seals or valve packing (except reactor coolant pump seal injection or leakoff) that is captured and conducted to collection systems or a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MEMBERS OF THE PUBLIC

1.17 MEMBERS OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose. R165

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8. R134 R169 R159

OPERABLE - OPERABILITY

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

SOLIDIFICATION

1.32 Deleted. R146

SOURCE CHECK

1.33 Deleted. R146

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of: R146

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.35 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. R146

UNIDENTIFIED LEAKAGE

1.36 UNIDENTIFIED LEAKAGE shall be all leakage (except reactor coolant pump seal water injection or leakoff) that is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.37 An UNRESTRICTED AREA shall be any area, at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional, and/or recreational purposes. R146 R63

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection instrumentation shall be OPERABLE:

a. Two lower containment atmosphere radioactivity monitors (gaseous and particulate), and

b. One containment pocket sump level monitor.

APPLICABILITY: MODES I, 2, 3 and 4.

ACTIONS:

a. With both containment pocket sump monitors inoperable, operation may continue for up to 30 days provided SR 4.4.6.2.1 is performed once per 24 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With either or both the gaseous or particulate lower containment atmosphere radioactivity monitors inoperable, operation may continue for up to 30 days provided grab samples of the lower containment atmosphere are analyzed once per 24 hours or SR 4.4.6.2.1 is performed once per 24 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

c. With both containment pocket sump monitors and both lower containment atmosphere radioactivity monitors inoperable, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection instrumentation shall be demonstrated OPERABLE by:

a. Performance of the lower containment atmosphere gaseous and particulate monitor CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Performance of containment pocket sump level monitor CHANNEL CALIBRATION at least once per 18 months.

* Surveillance performance not required until 12 hours after establishment of steady state operation.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day of primary-to-secondary leakage through any one steam generator, and |R213

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be verified to be within each of the above limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours.*

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

4.4.6.2.2 Verify steam generator tube integrity is in accordance with the requirements of Technical Specification 3/4.4.5, "Steam Generators."

* Not required to be performed until 12 hours after establishment of steady state operation.

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.3 Leakage from each Reactor Coolant System Pressure Isolation Valve, specified in Table 3.4-1, shall be equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure ≥ 2215 psig and ≤ 2255 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the residual heat removal system flow path when in, or during the transition to or from, the residual heat removal mode of operation.

ACTIONS:

a. With one or more flow paths with leakage from one or more Reactor Coolant System Pressure Isolation Valves greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual, deactivated automatic, or check valve* and restore the inoperable Reactor Coolant System Pressure Isolation Valve to OPERABLE status within the following 68 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Separate entry into the above ACTION is allowed for each flow path.

c. Entry into the applicable ACTIONS for systems made inoperable by an inoperable Reactor Coolant System Pressure Isolation Valve is required.

SURVEILLANCE REQUIREMENTS

4.4.6.3 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing requirements required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit#:

a. At least once per 18 months

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months.

c. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

Not required to be performed in MODES 3 and 4.

---

* Each valve used to satisfy ACTION a must have been verified to meet the Surveillance Requirement 4.4.6.3 and be in the reactor coolant pressure boundary.

\# Not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the Residual Heat Removal flow path when in the shutdown cooling mode of operation.

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.6 SEAL INJECTION FLOW

LIMITING CONDITION FOR OPERATION

3.5.6 Reactor coolant pump seal injection flow shall be within the limits of Figure 3.5.6-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

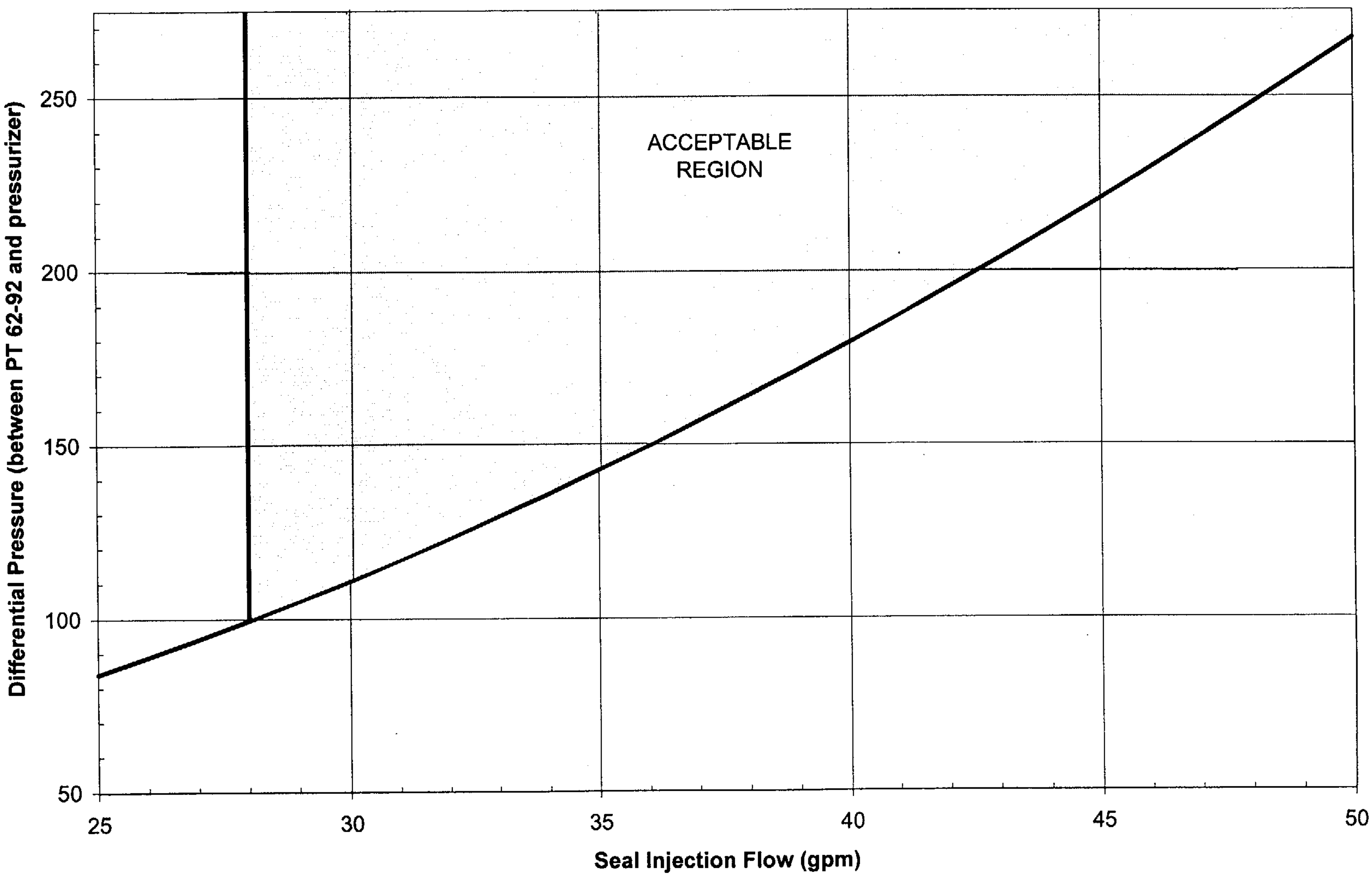
With reactor coolant pump seal injection flow not within limits, adjust manual seal injection throttle valves to give a flow within limit in accordance with Surveillance Requirement 4.5.6 within 4 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.6 At least once per 31 days* verify manual seal injection throttle valves are adjusted to give a flow within the emergency core cooling system safety analysis limits in Figure 3.5.6-1.

*This surveillance is not required to be performed until 4 hours after the reactor coolant system pressure stabilizes at $\geq$ 2215 psig and $\leq$ 2255 psig.

FIGURE 3.5.6-1
Seal Injection Flow Limits



3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION INSTRUMENTATION

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of reactor coolant system (RCS) leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all UNIDENTIFIED LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment pocket sump used to collect UNIDENTIFIED LEAKAGE is instrumented to alarm for increases of 1.0 gpm in the normal flow rates within one hour. This sensitivity is acceptable for detecting increases in UNIDENTIFIED LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} µCi/cc radioactivity for particulate monitoring and of 10^{-6} µCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS leakage.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS leakage.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer UNIDENTIFIED LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 3). Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS leakage into the containment area is necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public. Exclusions to the requirements of General Design Criteria 4, for the dynamic effects of the RCS piping, have been utilized based on the leak detection capability to identify leaks before a pipe break would occur.

RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS leakage indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, one containment pocket sump monitor, in combination with a gaseous and particulate radioactivity monitor, provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

Action a:

With both containment pocket sump monitors inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, Surveillance 4.4.6.2.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A footnote is added allowing that SR 4.4.6.2.1 is not required to be performed until 12 hours after establishing steady state operation (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup, letdown, and RCP seal injection and return flows). The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required pocket sump monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the frequency and adequacy of the RCS water inventory balance required by Action a.

Action a is modified by a note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

If the requirements of Action a cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 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.

Action b:

With either the gaseous or particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with Surveillance 4.4.6.2.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the containment atmosphere radioactivity monitors.

The 24 hour interval provides periodic information that is adequate to detect leakage. A footnote is added allowing that SR 4.4.6.2.1 is not required to be performed until 12 hours after establishing steady state operation (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup, letdown, and RCP seal injection and return flows). The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

Action b is modified by a note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous and particulate containment atmosphere radioactivity monitor channels are inoperable. This allowance is provided because other instrumentation is available to monitor for RCS leakage.

If the requirements of Action b cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 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.

Action c:

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown to a MODE in which the requirement does not apply is required. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 hours.

SURVEILLANCE REQUIREMENTS

Surveillance 4.4.6.1.a

This surveillance requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence that the monitors are operating properly. The frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

This surveillance requires the performance of a CHANNEL CALIBRATION for the required containment atmosphere radioactivity monitors. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Operating experience has proven that this frequency is acceptable.

This surveillance requires the performance of a CHANNEL FUNCTIONAL TEST on the required containment atmosphere radioactivity monitors. The test ensures that the monitors can perform their functions in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

The surveillance frequencies for these tests are specified in Table 4.3-3.

Surveillance 4.4.6.1.b

This surveillance requires the performance of a CHANNEL CALIBRATION for the required containment pocket sump monitors. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.

2. Regulatory Guide 1.45.

3. FSAR, Sections 5.2.7 "RCBP Leakage Detection Systems" and 12.2.4 "Airborne Radioactivity Monitoring."

3/4.4.6.2 OPERATIONAL LEAKAGE

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational leakage LCO is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified leakage is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary leakage as the initial condition.

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via safety valves for up to 30 minutes. Operator action is taken to isolate the affected steam generator within this time period. The 1 gpm primary to secondary leakage is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm primary to secondary leakage in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction
of these limits). Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 8.21 gpm at atmospheric conditions and 70°F in the faulted loop, which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines. If the projected and cycle distribution of crack indications results in primary-to-secondary leakage greater than 8.21 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 8.21 gpm. R227

The RCS operational leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the RCPB. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

b. UNIDENTIFIED LEAKAGE

One gpm of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment pocket

sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the leakage is from the pressure boundary.

c. Primary to Secondary Leakage through Any One Steam Generator (SG)

The 150 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

The 150-gallons per day limit incorporated into Surveillance 4.4.6.2.1 is more restrictive than the standard operating leakage limit and is intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that, should a significant leak be experienced, it will be detected, and the plant shut down in a timely manner. R227

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the RCS Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

APPLICABILITY In MODES 1, 2, 3, and 4, the potential for reactor coolant PRESSURE BOUNDARY LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

LCO 3/4.4.6.3, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS leakage when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.

ACTIONS

Action a:

If any PRESSURE BOUNDARY LEAKAGE exists, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within the following 30 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

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. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Action b:

UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or primary-to-secondary leakage in excess of the LCO limits must be reduced to within limits within 4 hours. This completion time allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB. If UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or primary to secondary leakage cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within the following 30 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

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. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

Surveillance 4.4.6.2.1

Verifying RCS leakage to be within the LCO limits ensures the integrity of the RCPB is maintained. PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of an RCS water inventory balance. Primary-to-secondary leakage is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup, letdown, and RCP seal injection and return flows). Therefore, a footnote is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Performance of this surveillance within the 12-hour allowance is required to maintain compliance with the provisions of Specification 4.0.3.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment pocket sump level. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3/4.4.6.1, "Leakage Detection Instrumentation."

The 72 hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

Surveillance 4.4.6.2.2

This surveillance provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this surveillance cannot be performed at normal operating conditions.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.4.3.

3/4.4.6.3 REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE LEAKAGE

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define reactor coolant system (RCS) pressure isolation valves (PIVs) as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment or an unanalyzed accident that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC, "Reactor Safety Study," (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System;

b. Safety Injection System; and

c. Chemical and Volume Control System.

The PIVs are listed in Table 3.4-1.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS PIV leakage is IDENTIFIED LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 6 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

Action a:

The flow path must be isolated. Action a is modified by a note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB.

Action a requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4-hour completion time allows the actions and restricts the operation with leaking isolation valves.

The 72-hour completion time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this action and the low probability of a second valve failing during this period.

If leakage cannot be reduced or the system isolated, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within the following 30 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. 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.

Action b:

Action b provides clarification that each flow path allows separate entry into Action a This is allowed based upon the functional independence of the flow path.

Action c:

Action c requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

SURVEILLANCE REQUIREMENTS

Surveillance 4.4.6.3

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Action a is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month frequency is consistent with 10 CFR 50.55a(g) (Ref. 7) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 6), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the surveillances were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this surveillance. The note that allows this provision is complementary to the frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. ASME, Boiler and Pressure Vessel Code, Section XI.
7. 10 CFR 50.55a(g).

3/4.5.6 SEAL INJECTION FLOW

BACKGROUND

The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during safety injection.

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

This LCO ensures that seal injection flow will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of the NRC Policy Statement.

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is established by adjusting the RCP seal injection needle valves to provide a total seal injection flow in the acceptable region of Technical Specification Figure 3.5.6-1. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The flow limits established by Technical Specification Figure 3.5.6-1 are consistent with the accident analysis.

The limits on seal injection flow must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTION

With the seal injection flow exceeding its limit, the amount of charging flow available to the RCS may be reduced. Under this condition, action must be taken to restore the flow to below its limit. The operator has 4 hours from the time the flow is known to be above the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The completion time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is conservative with respect to the completion times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.

When the actions cannot be completed within the required completion time, a controlled shutdown must be initiated. The completion time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown from MODE 3, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

Surveillance 4.5.6

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a flow within the limit ensures that proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. The differential pressure that is above the reference minimum value is established between the charging header (PT 62-92) and the RCS, and total seal injection flow is verified to be within the limits determined in accordance with the ECCS safety analysis (Ref. 3). The seal water injection flow limits are shown in Technical Specification Figure 3.5.6-1. The frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve surveillance frequencies. The frequency has proven to be acceptable through operating experience.

The requirements for charging flow vary widely according to plant status and configuration. When charging flow is adjusted, the positions of the air-operated valves, which control charging flow, are

adjusted to balance the flows through the charging header and through the seal injection header to ensure that the seal injection flow to the RCPs is maintained between 8 and 13 gpm per pump. The reference minimum differential pressure across the seal injection needle valves ensures that regardless of the varied settings of the charging flow control valves that are required to support optimum charging flow, a reference test condition can be established to ensure that flows across the needle valves are within the safety analysis. The values in the safety analysis for this reference set of conditions are calculated based on conditions during power operation and they are correlated to the minimum ECCS flow to be maintained under the most limiting accident conditions.

As noted, the surveillance is not required to be performed until 4 hours after the RCS pressure has stabilized within a ± 20 psig range of normal operating pressure. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the surveillance is timely. Performance of this surveillance within the 4-hour allowance is required to maintain compliance with the provisions of Specification 4.0.3.

REFERENCES

1. FSAR, Chapter 6.3 "Emergency Core Cooling System" and Chapter 15.0 "Accident Analysis".

2. 10 CFR 50.46.

3. Westinghouse Electric Company Calculation CN-FSE-99-48

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 259 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 250 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By letter dated June 30, 1999, as supplemented June 16, 2000, and August 3, 2000, the Tennessee Valley Authority (TVA, licensee) proposed a license amendment to change the Technical Specification (TS) for the Sequoyah Nuclear Plant (SQN), Units 1 and 2. TVA states that the purpose of its change is to make its TS more consistent with NUREG-1431, "Standard Technical Specifications for Westinghouse Plants." The proposed changes would revise TS Section 3/4.4.6, "Reactor Coolant System (RCS) Leakage," and its related definitions and basis, as well as relocate some of the requirements to different TS sections. The changes to TS Section 3.4.6.1, "RCS Leakage Detection System," include the title of this section, limiting conditions for operation (LCO), action statements, and surveillance requirements.

The June 16, 2000, and August 3, 2000, letters provided clarifying information and changes that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original notice.

2.0 BACKGROUND

The purpose of the proposed change to the SQN TS is to provide requirements that are more consistent with the industry standard in accordance with NUREG-1431. TVA is utilizing this revision to relocate requirements to new specifications that provide better implementation and are fashioned to address the appropriate safety systems. These relocations are consistent with NUREG-1431 and subsequent U.S. Nuclear Regulatory Commission (NRC) approved changes (i.e., TS Traveler Form (TSTF) 54, Revision 1; TSTF-60, Revision 0; and TSTF-116, Revision 2). TVA is deleting items, in accordance with NUREG-1431, that are not necessary for maintaining operability of these systems or are adequately covered by requirements in other specifications. These revisions also incorporate modifications for the LCOs, applicability, action times, and Surveillance Requirements (SRs) that are acceptable based on the SQN design and NUREG-1431 recommendations. For the RCS leakage detection specification, this change will eliminate the potential to initiate an unnecessary unit shutdown when sufficient leak detection capability is available. TVA has added expanded Bases to complete this effort to improve consistency with NUREG-1431 for these specifications.

Components of the SQN leakage detection system include containment radiation monitors, humidity monitors, a reactor vessel flange leakoff temperature detector, condenser vacuum pump radiation monitors, component cooling system radiation monitors, charging pump flow indicators, reactor building containment floor and equipment drain sump level monitors, main steam line radiation monitors, and safety valve and power-operated relief valve leak detection and valve position monitors. These systems provide a means of detecting, to the extent practical, leakage from the reactor coolant pressure boundary. Built-in redundancy and diversity is a key factor in the system. Various types of detectors serve to supplement one another since the range of each detector either overlaps or duplicates the range of other detectors. Detector sensitivities are such that they provide the capability to sense a leak well before the leakage becomes unacceptable.

Using several types of detectors with various sensitivities results in a system more than adequate to detect abnormal leakage. Multiple types of sensors assure early leak detection in case of failure of one or more types, thereby assuring that the necessary margin of safety will not be exceeded. The monitors credited in the accident analysis and required by the SQN TSs are the containment radiation monitors and the reactor building containment floor and equipment drain sump level monitors.

During plant life, the component joint and valve interfaces can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The purpose of the RCS operational leakage LCO in TSs is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the identified leakage from the unidentified leakage is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur.

RCS pressure isolation valves are defined as any two normally-closed valves in series within the reactor coolant pressure boundary, which separate the high pressure RCS from an attached low pressure system. Over time, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS pressure isolation valve leakage requirements in the TSs allow RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

Although the TS requirement provides a limit on allowable pressure isolation valve leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the pressure isolation valves between the RCS and the connecting systems are degraded or degrading. Pressure isolation valve leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss-of-coolant accident outside containment that could degrade the ability for low pressure injection.

The function of the seal injection throttle valves during an accident is similar to the function of the emergency core cooling system (ECCS) throttle valves in that each restricts flow from the centrifugal charging pump header to the RCS. The restriction on reactor coolant pump seal

injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because reactor coolant pump seal injection flow is not isolated during safety injection.

TVA proposes to revise TS Section 3.4.6.2, "RCS Operational Leakage," to be consistent with NUREG-1431. This revision relocates the requirements for controlled leakage (seal injection flow) and pressure isolation valve leakage to two new specifications (3.4.6.3 and 3.5.6). In conjunction with this change, the definitions in Section 1.0 of the TSs would be revised accordingly. This involves the deletion of the definition for controlled leakage and modification of the identified leakage and unidentified leakage definitions to address controlled leakage as seal injection flow. Other minor editorial enhancements, as provided in NUREG-1431, have been proposed. The NUREG-1431 provision for total primary to secondary leakage through all steam generators (S/Gs) was not included because of recent NRC-approved SQN License Amendments 222 and 213, dated April 9, 1997, for Units 1 and 2, respectively, that have justified this omission. Section 3.4.6.2 actions have been revised to accommodate the removal of the pressure isolation valves from this specification. Otherwise, the actions are unchanged and are consistent with NUREG-1431. Several of the current surveillance requirements (SRs) have been deleted because either they would be redundant to the controlled leakage and pressure isolation valve TS relocation (TS3.4.6.1) or they would not be consistent with recommended requirements in NUREG-1431. The deleted SRs include SR 4.4.6.2.1.a, SR 4.4.6.2.1.b, SR 4.4.6.2.1.c, and SR 4.4.6.2.1.e. The deleted SRs for the NUREG include those that are associated with leak detection and the reactor head flange leakoff. An SR has been added to address the S/G tube integrity verifications, and a footnote has been incorporated to describe the appropriate time to perform the RCS inventory balance such that accurate results will be accomplished. In order to complete the relocation of the pressure isolation valve requirements, Table 3.4-1 that lists these valves has been moved to the new Specification 3.4.6.3.

New Specification 3.4.6.3 is being added to provide the requirements for the RCS pressure isolation valves. The operability requirements along with the applicability, actions, and SRs have been developed consistent with NUREG-1431. Those portions of NUREG-1431 that are not applicable to the SQN design or that provide a relaxation or exclusion that is not appropriate have not been included in the proposed request.

The LCO requirements have been modified to utilize the current industry criteria for leakage limits of less than or equal to 0.5 gallons per minute (gpm) per nominal inch of valve size up to a maximum of 5 gpm. The applicability requirements have been revised to provide an exception to the residual heat removal (RHR) system flow path in Mode 4 when in or during transition to or from RHR operation. The actions have been modified to allow a limited time of operation with valve leakage above this limit, provided the flow path is isolated. This isolation must be with a valve that has been tested and meets the same leakage requirements of this specification, as well as being within the reactor coolant boundary. Additional action requirements have been included for clarity involving the ability to separately enter this action for each inoperable flow path and the need to enter applicable actions for systems rendered inoperable by the inoperable RCS pressure isolation valve. The surveillance testing requirements are not changed by this request; however, the current criteria that specifically requires the performance of the surveillance following maintenance, repair or replacement work on the valve, has been removed; and the duration in cold shutdown before requiring performance has been extended to 7 days. The changes to SQN's current requirements

described above are consistent with NUREG-1431. TVA has included a listing of pressure isolation valves associated with this specification that contains the same information found in the current operational leakage specification.

New TS 3.5.6 is being added to provide the requirements for the seal injection flow. This specification takes the place of the current requirements in the operational leakage specification for controlled leakage. The new specification is fashioned after NUREG-1431; however, a modification to the NUREG has been incorporated that utilizes a relationship involving the pressure differential between the charging pump discharge header and the RCS and total seal injection flow. This method for determining acceptable seal injection flows does not require the current provision to have the modulating valve fully open in consideration of the SQN design for the location of the header pressure sensor. The current flow limit of 40 gpm at normal RCS pressure is replaced by range of flow limits that correspond to this amount of flow restriction over various amounts of pressure differential between the header and the RCS.

The surveillance for Specification 3.5.6 will utilize this range of differential pressures and associated flow values to verify acceptable settings of the seal injection throttle valves. The figure that contains these relationships will be included in the new specification. A footnote, consistent with NUREG-1431, has been added to the surveillance that clarifies the need to be at normal RCS pressure and stabilized for 4 hours prior to the performance of the flow verification. This provision replaces the existing exception for Specification 4.0.4. This specification will apply in Modes 1, 2, and 3, which is a modification of the current requirement that includes Mode 4. As a result of this change, the action will only require a shutdown to Mode 4 within a total of 12 hours if inoperable seal injection flows cannot be corrected.

TVA is adding expanded Bases for each specification revised in this proposed change. These Bases have been developed in accordance with the recommendations of NUREG-1431. Modifications and omissions have been utilized where necessary to agree with the specific requirements proposed and the SQN design. The proposed Bases for Specifications 3.4.6.1 and 3.4.6.2 supersede the current discussions. TVA has also proposed the necessary revisions to the TS index to support the specification and Bases changes described above.

3.0 EVALUATION

Licensee Safety Analysis

TVA provided the following discussion of safety impact of these changes:

> The majority of the revisions proposed in this request for RCS leakage detection, RCS operational leakage, RCS pressure isolation valve leakage, and ECCS seal injection flow do not alter the intent or the application of the current TS requirements. The purpose for these revisions is to enhance consistency with NUREG-1431. Other revisions are proposed that provide more reasonable requirements that are also consistent with NUREG-1431, but provide some flexibility to the current TS requirements. One revision modifies NUREG-1431 recommendations consistent with an amendment approved by NRC for another licensee. The following discussions provide the specific impact for the revisions to each specification.

The revisions to TS 3.4.6, "Leakage Detection Instrumentation," title and LCO are editorial changes that incorporate descriptions similar to NUREG-1431. Combining the lower containment radiation monitors into a single item is utilized to group the two types of leak detection instrumentation together, which simplifies the application of action requirements. These changes do not impact safety because they continue to require the same diverse components for RCS leak detection purposes that can rapidly detect small leaks.

The actions have been replaced by NUREG-1431 recommendations for leak detection instrumentation. These revisions implement new requirements for the complete loss of either radiation or sump level monitoring for RCS leak detection as well as loss of both functions. These actions provide more flexibility because the current action did not allow for the complete loss of the containment radiation monitors. This change is acceptable because the sump level monitoring function, along with the additional sampling requirements, continue to ensure that small RCS leaks can be detected rapidly for a limited period of time. This change also eliminates the potential to require an unnecessary unit shutdown when sufficient leak detection is available as required by Action b. The proposed Action a, for the loss of the sump level monitoring, does not alter the current requirements except for the replacement of the requirement to use containment grab samples. The requirement to take an RCS inventory balance in place of the grab samples, provides a more diverse indication of leakage in place of the inoperable sump level monitors. The use of a grab sample would only provide compensatory monitoring equivalent to the remaining radiation monitor functions. Performance of the inventory balance or grab samples is allowed for inoperable radiation monitors in Action b because either action will provide a diverse method of leak detection and supplement the capability of the sump level monitors to detect leakage. A footnote is included for Actions a and b to allow the performance of an RCS inventory balance to be within 12 hours of steady state operation to ensure accurate results. The expectation is that the RCS balance can be accomplished with proper results within the 12 hours following steady state operations.

The frequency for these actions and the duration that these actions can be relied upon has not been changed. Consistent with NUREG-1431, an exception to TS 3.0.4 is provided for Actions a and b to allow mode changes with inoperable leak detection monitors because other instrumentation is available to monitor RCS leakage. The shutdown requirements, for failure to be able to comply with the actions, have not been changed. The current immediate shutdown action for the loss of all leak detection instrumentation has been retained in the specific shutdown requirement of Action c. This requirement requires shutdown to Mode 3 within 6 hours and Mode 5 within the following 30 hours. This action is more conservative than the NUREG-1431 recommendation that utilizes TS 3.0.3 for the loss of all leak detection because the initial 1-hour provision allowed prior to initiating shutdown is not utilized.

The SRs have not been changed in intent, but the wording has been modified to agree with NUREG-1431 and the changes in the LCO. These changes to Specification 3.4.6.1 will continue to provide acceptable RCS leak detection functions and appropriate actions for the inoperability of either type or both detection functions. These changes will not adversely impact nuclear safety.

The revisions to the LCO for TS 3.4.6.2, "Operational Leakage," involve the relocation of the controlled leakage requirement to new Specification 3.5.6, "Seal Injection Flow," and the pressure isolation valve requirement to new Specification 3.4.6.3, "Reactor Coolant System Pressure Isolation Valves." These relocations do not eliminate the associated requirements, but place them in specifications that are more appropriate and consistent with NUREG-1431. The NUREG-1431 provision for primary-to-secondary leakage through all S/Gs has not been incorporated. This is based on recent NRC approved SQN License Amendments 222 and 213 for Units 1 and 2, respectively, that eliminated this provision in conjunction with S/G [steam generator] tube inspection and plugging methods and NRC Generic Letter 95-05. The controlled leakage definition is deleted in TS Section 1.8 and is now referred to as seal injection flow. Other definitions in Section 1.16 for identified leakage and Section 1.36 for unidentified leakage have been modified to agree with the wording in NUREG-1431 and the replacement of controlled leakage with seal injection flow. There is no change in the intent of these definitions; only the term for controlled leakage has been modified and clarifications have been incorporated as recommended in NUREG-1431.

The actions for operational leakage have not been changed. Those actions that applied to pressure isolation valves have been relocated to Specification 3.4.6.3. No change was necessary to these actions to accommodate the relocation of the seal injection flow requirements.

The SRs have been significantly revised based on the relocated portions of this specification and the recommendations of NUREG-1431. For the surveillances that are proposed for operational leakage, minor wording changes have been incorporated consistent with the NUREG, and new Surveillance 4.4.6.2.2 has been added to specifically address the primary-to-secondary leakage requirements. This new surveillance duplicates the purpose of TS 3.4.5 and is being added to provide consistency with NUREG-1431 and future modification of the SQN TSs in accordance with the standard. A footnote has been incorporated for the performance of the RCS inventory balance in Surveillance 4.4.6.2.1 that supports the performance during steady state operation and within 12 hours after establishment of steady state operation. This provision ensures that the RCS conditions are acceptable to properly obtain valid inventory balance results. Inventory balance calculations during maneuvering of the RCS parameters do not provide useful information regarding RCS leakage. The specific requirements for steady state operation is provided in the Bases for this specification.

The current Surveillances [SRs] 4.4.6.2.1.a and 4.4.6.2.1.b have been deleted because the requirements for the operability of these functions are ensured

by Specification 3.4.6.1 for leakage detection. These SRs are redundant to Specification 3.4.6.1 and do not provide additional benefit. Surveillance 4.4.6.2.1.c has been relocated to the seal injection flow specification. The deletion of Surveillance 4.4.6.2.1.e for monitoring reactor head flange leakoff is consistent with NUREG-1431. This surveillance does not provide a significant improvement in the assessment of RCS leakage in comparison to the RCS inventory balance. The leakage detection function is also fully satisfied without the use of this design feature. While this feature does not need to be controlled by the TSs, information from this system will continue to be used for the identification of leakage and its source. Surveillance 4.4.6.2.2 for RCS pressure isolation valves has been relocated to Specification 3.4.6.3.

The proposed changes to the operational leakage requirements in Specification 3.4.6.2 will continue to provide acceptable attributes to ensure that RCS leakage is properly identified and appropriate actions are taken to minimize the impact of RCS leaks. The portions that have been relocated to new specifications will contain appropriate controls to maintain acceptable operability conditions for pressure isolation valves and seal injection flow. Therefore, the proposed revisions to Specification 3.4.6.2 will not adversely impact nuclear safety.

TVA proposes new Specification 3.4.6.3, "Reactor Coolant System Pressure Isolation Valves," that will provide requirements previously contained in the RCS operational leakage section of the TSs. This relocation is proposed for consistency with NUREG-1431. TVA has revised the LCO for the pressure isolation valves consistent with the latest industry standard and recommendations in NUREG-1431.

The proposed LCO requires the pressure isolation valve leakage to be less than or equal to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm. The LCO continues to require the RCS to be at nominal operating pressure of 2235 pounds per square inch gauge (psig) with a tolerance of plus or minus 20 psig. The leakage requirement provides a relaxation of the current 1 gpm limit that imposed an unjustified penalty on the larger valves without providing information on potential valve degradation. The 1 gpm limit also has the potential to increase personnel radiation exposures because the time to perform surveillance testing could be greater because of increased work activities in radiation fields. The revision to a valve size related leakage criteria is acceptable because associated systems that have larger valves also have greater pressure relief capability. The new criteria allows for leakage above 1 gpm, although limited to a maximum of 5 gpm, because the isolated low pressure systems will not be overpressurized based on their relief capacity being greater than the allowed leakage limit. Therefore, the proposed change to the LCO will result in lower radiation exposures to personnel and a superior leak rate limit based on valve size as compared to a single allowable value.

The proposed applicability for pressure isolation valves has been modified for Mode 4 to provide an exception for valves in the RHR flow path when in or

during transition to or from the RHR operation. This change ensures the acceptability to operate the RHR system in Mode 4 when necessary to perform RCS heat removal. This clarification enhances the applicability of this specification that was not intended to restrict the use of systems that remove decay heat and provides consistency with NUREG-1431.

The actions for pressure isolation valves have been modified consistent with NUREG-1431. Action a incorporates a provision that allows pressure isolation valve leakage to be in excess of the limit for up to a total of 72 hours provided sufficient isolation is achieved within the initial 4 hours of inoperability. This isolation must be achieved by a valve that has been acceptably tested to the same leakage requirements as the pressure isolation valves and be within the RCS pressure boundary. The 72-hour completion time after exceeding the limit provides a reasonable interval to restore the valve to operable status. This time frame considers required activities to complete the action and the low probability of a second valve failing during this period. This change will not adversely impact nuclear safety because the flow path will be sufficiently isolated at all times, and the period of time without redundant isolation capability will be appropriately limited. The allowance in NUREG-1431 to isolate the flow path with two valves within 72 hours and be permitted to operate indefinitely has not been incorporated. This is based on the SQN design that does not have three or more valves within the RCS pressure boundary that could perform this capability. The addition of this provision would not be appropriate because it could not be achieved.

Two additional actions have been added to the pressure isolation valve requirements that provide clarifications for the proper application of Action a. Action b clarifies that the actions can be entered separately for each flow path based on the functional independence of the flow paths. Action c requires entry into the actions for systems rendered inoperable as a result of the inoperable pressure isolation valve. This action requires an evaluation of systems that could be affected to ensure that the leakage has not impacted the safety function of those systems. These footnotes do not alter the application of the pressure isolation valve requirements, but serve to enhance the understanding of the expectations for applying these requirements. The NUREG-1431 recommendation to include an action for the auto-closure interlock for the RHR system has not been incorporated. This is based on NRC approved TS Amendments 139 and 128 for Units 1 and 2, respectively, in 1990. This amendment approved the removal of this feature because spurious actuations during periods when RHR was required created a greater safety concern.

The SRs for the pressure isolation valves have been retained without significant modification and are consistent with NUREG-1431. The previously described exception to the TS applicability when RHR is in service is reiterated in a footnote to this surveillance to clarify that the surveillance does not apply to associated valves under this condition. Item b of the surveillance has been modified consistent with NUREG-1431 to require performance of the test if the unit has been in Mode 5 for 7 or more days. This is an

extension of the current 72-hour interval that will minimize unnecessary testing for short duration unit outages. In addition, a reduction in radiation exposure to testing personnel will result by eliminating testing activities that do not significantly enhance the function of the pressure isolation valves. This change is consistent with the recommendations provided in NUREG-1366. The current requirement to test the valves following maintenance has been omitted based on SQN maintenance practices, which are further enhanced by the implementation of the Maintenance Rule. A similar deletion to the SQN TSs was approved on June 13, 1995, by NRC in Amendments 203 and 193 for Units 1 and 2, respectively, where testing was required following maintenance activities. This deletion is acceptable because maintenance practices ensure that appropriate testing is conducted following maintenance and a specific TS requirement is not necessary.

Consistent with NUREG-1431, the exception to Specification 4.0.4 has been revised to indicate that the surveillances are not required to be met in Modes 3 and 4. This provides the proper conditions to perform the surveillance and ensures that Modes 1 and 2 are not entered unless the surveillance is current. The delay in testing the pressure isolation valves is based upon ensuring the proper conditions exist to perform the testing and assumes that any leakage that may exist prior to the testing is minimal such that RCS operating pressure can be achieved without operational impact. In regards to the NUREG-1431 recommended note that minimizes repetitive testing of valves, TVA has elected to not incorporate this exception at this time as the design and testing methodology preclude the actuation of the pressure isolation valves that have already been tested. The proposed surveillance for the RCS pressure isolation valves and the criteria for performance provide acceptable requirements to ensure that the leakage from these valves will not impact their operability or nuclear safety.

The proposed changes to Specification 3.4.6.3 for the RCS pressure isolation valves will continue to ensure that excessive leakage through these valves is properly identified and resolved. The surveillance will provide the appropriate test to detect leakage in excess of the established limits. When these limits are exceeded, the required actions will initiate appropriate activities to minimize the impact of the leakage. While the LCO requirement has been modified to be more reasonably based on valve size, it will continue to provide a limit that will maintain nuclear safety. Considering the SQN design features, these changes are consistent with NUREG-1431 and will not result in an adverse impact to nuclear safety.

TVA proposes new Specification 3.5.6, "Seal Injection Flow," in the ECCS portion of the TS. These requirements have been relocated from the operational leakage requirements in the RCS portion of TS. Placing this new specification in the ECCS section of TS is appropriate because the seal injection flow limit is intended to ensure that ECCS flow is not diverted to a level that the safety function is degraded. The proposed LCO for this specification has been developed consistent with an NRC approved specification for the Vogtle Electric Generating Plant during their conversion

to the NUREG-1431 recommendations. The proposed LCO utilizes a range of differential pressures between the charging header and the RCS and the associated acceptable seal injection flows that will support accident analysis assumptions. This relationship was evaluated and developed by Westinghouse Electric Company based on SQN specific piping configurations and is depicted in a new Figure 3.5.6-1. This approach will provide flow limits that will ensure that seal injection flows do not exceed the amounts that could degrade the ECCS function. The proposed limits are equivalent to the current 40 gpm limit, but will normally provide the ability to utilize existing plant conditions for this verification without manipulating charging header valve positions. The current requirements do not prescribe sufficient parameters to ensure that appropriate conditions are established to verify the seal injection flow resistance. The proposed revision will enhance this SR by establishing a differential pressure and the corresponding flow limit.

The applicability for this specification has been revised to delete the Mode 4 consideration and to provide consistency with NUREG-1431. This change is based on seal injection flow in Mode 4 and lower, where RCS pressure is lower and decay heat removal requirements are reduced, not being as critical such that ECCS functions would be impacted. As a result, the proposed action for unacceptable seal injection flow conditions has not changed with the exception that shutdown requirements for not complying with the actions only require a mode reduction to hot shutdown rather than cold shutdown. This change is acceptable because the ECCS functions will not be adversely impacted during Mode 4 or lower operation considering seal injection flow.

The SR for seal injection flow remains the same with only the limits of the LCO being altered. The RCS pressure requirement for this surveillance has been clarified to require stabilized conditions at normal RCS operating pressure for at least 4 hours prior to performance. This provision ensures that the RCS conditions are representative and sufficiently stabilized to provide accurate flow verifications and the 4-hour provision will support timely surveillance performance. This footnote continues to satisfy the existing exclusion for Specification 4.0.4 such that entry into Mode 3 is permitted in order to establish the appropriate testing conditions. In addition, the position of the modulating valve is not specified because the differential pressure requirements in the new limits are independent of the valve position. This is because the valve is upstream of the charging header pressure detector, and its position will not alter the measured differential pressure for the seal injection flow limit.

The proposed specification for the seal injection flow will provide a more appropriate set of requirements to ensure that the ECCS safety functions are not impacted. The proposed requirements provide flexibility in system conditions for the verification of acceptable flow values. This flexibility does not reduce ECCS function confidence because the same criteria is utilized in an equivalent method. This provision has been developed through an analysis of the SQN system design to ensure the same level of protection is maintained. These revisions are consistent with the intent of the

NUREG-1431 recommendations with a modification to the limits that have been previously approved by NRC. Therefore, the proposed specification for the seal injection flow will continue to provide appropriate requirements to maintain acceptable flows and ECCS operability.

TVA has also proposed expanded Bases for each of the TS sections described. These Bases have been developed from the Bases provided in NUREG-1431.

NRC Staff Evaluation

TS 3.4.6.1 - Leakage Detection Instrumentation

The change to the title from "Leakage Detection System" to "Leakage Detection Instrumentation" is an editorial change that is consistent with NUREG-1431. Therefore, it is acceptable.

The original proposed change (June 30, 1999 letter) to LCO 3.4.6.1 combined LCO 3.4.6.1.a and LCO 3.4.6.1.c into one revised LCO 3.4.6.1.a. The revised proposed change in the TVA letter of June 16, 2000, was the result of a conference call on June 6, 2000, between TVA and the NRC staff. The original proposed TS change would have reduced the required number of operable leak detection instruments from three to two. This would not have been consistent with Regulatory Guide (RG) 1.45 Regulatory Position c.3, which is a current licensing basis for SQN. Position c.3 states that at least three separate detection methods should be employed for RCS leakage detection. The staff finds the revised form of LCO 3.4.6.1, dated June 16, 2000, consistent with NUREG-1431 and RG 1.45 and, therefore, acceptable.

The staff finds that the changes to the surveillance requirements in SR 4.4.6.1 are editorial in nature, do not change the context or content of the SR, and are, therefore, acceptable. The staff finds that the clarifying footnote added to the actions (when SR 4.4.6.2.1 is performed) to establish a 12-hour steady-state operation period before obtaining RCS inventory balance calculations is acceptable because inventory balance calculations during maneuvering of the RCS parameters do not provide useful information regarding RCS leakage. This footnote already exists for SR 4.4.6.2.1. The addition of this footnote is also consistent with NUREG-1431. Therefore, these changes are acceptable.

The staff reviewed the proposed changes to action statements and the associated completion time for LCO 3.4.6.1, and finds these changes to be consistent with NUREG-1431 and, therefore, acceptable.

TS 3.4.6.2 - Reactor Coolant System Operational Leakage and TS 3.5.6 - Seal Injection Flow

The current TS 3.4.6.2, "Reactor Coolant System Operational Leakage," specifies a maximum controlled leakage of 40 gpm at an RCS pressure of 2235 psig. The basis of this TS describes that the controlled leakage is the total flow delivered to the reactor coolant pump (RCP) seals. The RCP seal injection flow is supplied from the charging pumps, which are a part of the ECCS. The purpose of this limit is to assure that, in the event of a loss of coolant accident (LOCA), the ECCS flow will not be less than that assumed in the accident analyses. TVA proposed changes that would replace the fixed-value of the controlled leakage in TS 3.4.6.2

with a new TS 3.5.6, "Seal Injection Flow," to provide the restrictions for the seal injection flow, with a variable differential-pressure value in the form of a figure depicting an acceptable operating regime for seal injection flow.

The charging pumps are a part of the ECCS and are credited in the large and small LOCA accident analyses. The LOCA analyses establish the minimum flow requirements from the charging pumps for accident mitigation. The RCP seal injection flows are also supplied from the charging pump discharge and these flow paths are not isolated during a LOCA. Therefore, it is essential to restrict the maximum RCP seal injection flow to assure sufficient ECCS flow as assumed in the accident analyses.

New TS 3.5.6 is essentially structured consistent with NUREG-1431. However, the current RCP seal injection flow limit of 40 GPM at normal RCS operating pressure is replaced by a range of flow limits that corresponds to the amount of flow restriction over various amounts of pressure differential between the charging pump discharge header and the RCS. This approach will ensure that RCP seal injection flows do not exceed the amounts that could degrade the ECCS function as well as provide convenience for surveillance tests. The licensee-proposed RCP seal injection flow restrictions were evaluated by Westinghouse Electric Company based on the data from the SQN safety analyses and ECCS piping configurations. The change proposed by TVA places the RCP seal injection flow rate limits in new TS Figure 3.5.6-1. This would assure that the restrictions provided for the RCP seal injection flow will support ECCS performance assumed in accident analyses.

The staff has reviewed the proposed RCP seal injection flow limits (Figure 3.5.6-1) and concludes that these limits would provide a more conservative level of protection with respect to ECCS performance since they are developed based on the current TS value at normal operating RCS pressure. The staff has also reviewed the proposed TS 3.5.6 for RCP seal injection flow and concluded that the proposed changes are consistent with NUREG-1431, including the recommended allowable action times and surveillance intervals. The current TS requirements do not prescribe sufficient parameters to ensure that appropriate conditions are established to verify the seal injection flow resistance. The proposed revision will enhance this SR by establishing an RCP seal injection flow limit based upon the differential pressure between the charging header and the RCS. The associated SR (4.5.6) requires flow rate verification every 31 days to ensure that flow rate is within ECCS safety analysis limits. Hence the TS requirements will continue to maintain ECCS performance within the envelope of the safety analysis. Therefore, this change is acceptable because the current limits for RCP seal injection flow will continue to be maintained in accordance with existing requirements based on the safety analyses and plant operating procedures.

TS 3.4.6.3 - Pressure Isolation Valve (PIV) Leakage

The proposed changes would revise TS Section 3.4.6.2, relocate the requirements for PIV leakage to a new TS Section 3.4.6.3, and add a new Bases Section 3/4.4.6.3 for PIV leakage.

The existing TS 3.4.6.2 provides requirements for the maximum allowable leakage in PIVs including the limiting condition for operation, action requirements and surveillance requirements. TS 3.4.6.2(f) limits PIV leakage to:

> "1 GPM leakage at a Reactor Coolant System pressure .... from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4.1."

In early 1980, NRC revised the PIV leakage requirement from a single amount of 1 gpm to 0.5 gpm per nominal inch of valve size with a maximum of 5 gpm at reactor coolant system pressure. This change tightened the leakage requirement for smaller valves and relaxed it for larger ones. This revised PIV leakage requirement has been incorporated into NUREG-1431.

TVA states that the operability requirements, along with the applicability, actions, and SRs have been developed consistent with the recommended contents of NUREG-1431. In summary, the proposed changes would revise existing TS Section 3.4.6.2, relocate the requirements for PIV leakage to a new TS Section 3.4.6.3, and add a new Bases Section 3/4.4.6.3 for PIV leakage. The staff reviewed the proposed TS changes for PIV leakage against the recommendations contained in NUREG-1431. The staff finds that the format of the proposed changes is different, but the changes are consistent with the recommendations in NUREG-1431. Therefore, the staff finds that the proposed TS changes are acceptable.

The NRC staff concludes that the proposed TS changes for PIV leakage are acceptable on the basis that they are consistent with the recommendations in NUREG-1431,which were the subject of detailed NRC evaluation prior to issuance of the NUREG, and will reduce unnecessary occupational radiation exposure by eliminating PIV testing activities that do not significantly enhance the function of the PIVs. The enhanced TS will continue to assure that excessive PIV leakage is properly identified and resolved.

Definitions and Bases

The controlled leakage definition is deleted in TS Section 1.8 and is now referred to as seal injection flow, for reasons previously discussed. Other definitions in Section 1.16 for identified leakage and Section 1.36 for unidentified leakage have been modified to be consistent with the wording in NUREG-1431. Therefore, these definition changes are acceptable.

Conclusion

Based on the above evaluation, the NRC staff finds that the TS changes proposed by TVA in their request dated June 30, 1999, as amended on June 16, 2000, are acceptable on the basis that they are consistent with the recommendations in NUREG-1431. The staff has concluded that the proposed changes to and reorganization of the TS and Bases will enhance plant safety by reducing operator misunderstanding and misapplication of the TS. The revised Bases will provide additional details regarding the intent of the specifications and will help to assure the appropriate application of the TS requirements intended to detect and minimize excessive RCS leakage. Adaptation of several Improved Standard TS features in RCS leakage and leakage detection are judged by the staff to be an improvement in the SQN TS and, therefore, an improvement in plant operational safety.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in Title 10, Code of Federal Regulations, Part 20, and changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (64 FR 56533). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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