No Changes. Provided for Continuity Only.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.31	0.81	1.44
1050	0.32	0.86	1.57

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Surveillance Requirement 4.1.3.2.a or b, operation may continue provided that:

1. For all "slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, the individual scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.38	1.09	2.09
1050	0.39	1.14	2.22

2. For "fast" control rods, i.e., those which satisfy the limits of Specification 3.1.3.2, the average scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Average Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.30	0.78	1.40
1050	0.31	0.84	1.53

*For intermediate reactor vessel dome pressure, the scram time criteria are determined by linear interpolation at each notch position.

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No Changes. Provided for Continuity Only.

LIMITING CONDITION FOR OPERATION (Continued)

3.1.3.2 ACTION (Continued):

3. The sum of "fast" control rods with individual scram insertion times in excess of the limits of ACTION a.2 and of "slow" control rods does not exceed 5.

4. No "slow" control rod, "fast" control rod with individual scram insertion time in excess of the limits of ACTION a.2, or otherwise inoperable control rod occupy adjacent locations in any direction, including the diagonal, to another such control rod.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. With a "slow" control rod(s) not satisfying ACTION a.1, above:

1. Declare the "slow" control rod(s) inoperable and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more "slow" control rods declared inoperable.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

c. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Specification 4.1.3.2.c, operation may continue provided that:

1. "Slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, do not make up more than 20% of the 10% sample of control rods tested.

2. Each of these "slow" control rods satisfies the limits of ACTION a.1.

3. The eight adjacent control rods surrounding each "slow" control rod are:

   a) Demonstrated through measurement within 12 hours to satisfy the maximum scram insertion time limits of Specification 3.1.3.2 and

   b) OPERABLE.

4. The total number of "slow" control rods as determined by Specification 4.1.3.2.c, when added to the sum of ACTION a.3 as determined by Specification 4.1.3.2.a and b, does not exceed 5.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

~~b. For specifically affected individual control rods* following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and~~

b. ~~c.~~ For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

4.1.3.3 Insert

*The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 provided this surveillance requirement is completed prior to entry into OPERATIONAL CONDITION 1.

Insert for page 3/4 1-8

4.1.3.3 The maximum scram 'nsertion time for specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig.* Alternatively, those specific control rods may be determined OPERABLE with reactor coolant pressure less than 950 psig by demonstrating an acceptable scram insertion time to notch position 13. The scram time acceptance criteria for this alternate test shall be determined by linear interpolation between 0.95 seconds at a reactor coolant pressure of 0 psig and 1.40 seconds at 950 psig. If this alternate test is utilized, the individual scram time shall also be measured with reactor coolant pressure greater than 950 psig prior to exceeding 40% of RATED THERMAL POWER. For each of the above single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators.

No Changes. Provided for Continuity Only.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least six of the following valves and the relief valve function of at least five additional valves, other than those satisfying the safety valve function requirement, shall be OPERABLE with the specified lift settings; and the acoustic monitor for each OPERABLE valve shall be OPERABLE.*

Number of Valves	Function	Setpoint** (psig)
7	Safety	1165 ± 11.6 psi
5	Safety	1180 ± 11.8 psi
4	Safety	1190 ± 11.9 psi
1	Relief	1103 ± 15.0 psi
8	Relief	1113 ± 15.0 psi
7	Relief	1123 ± 15.0 psi

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety and/or relief valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck open safety/relief valve(s); if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve acoustic monitor(s) inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With either relief valve function pressure actuation trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within 7 days; otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

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* One relief valve pressure actuation channel and/or one acoustic monitor channel may be placed in an inoperable status for up to 2 hours for the purpose of performing surveillance testing in accordance with Specifications 4.4.2.1.1 and 4.4.2.1.2.

** The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

REACTOR COOLANT SYSTEM

SAFETY/RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months.*

4.4.2.1.2 The relief valve function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per 31 days.

b. CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST at least once per 18 months. Each of the two trip systems or divisions of the relief valve function actuation logic associated with the Nuclear System Protection System shall be manually tested independent of the SELF TEST SYSTEM during separate refueling outages such that both divisions and all channel trips are tested at least once every four fuel cycles.**

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*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam ~~pressure is~~ Conditions are adequate to perform the test.

**Manual testing for the purpose of satisfying Specification 4.4.2.1.2.b. is not required until after shutdown during the first regularly scheduled refueling outage.

No Changes. Provided for Continuity Only.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 ECCS Divisions I, II and III shall be OPERABLE with:

a. ECCS Division I consisting of:

1. The OPERABLE low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

3. Seven OPERABLE ADS valves.

b. ECCS Division II consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

2. Seven OPERABLE ADS valves.

c. ECCS Division III consisting of the OPERABLE high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*,#, and 3*,##

ACTION:

a. For ECCS Division I, provided that ECCS Divisions II and III are OPERABLE:

1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.

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*The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

#See Special Test Exception 3.10.5.

##One LPCI subsystem of the RHR system may be aligned in the shutdown cooling mode when reactor vessel pressure is less than the LPCI cut-in permissive setpoint.

No Changes. Provided for Continuity Only.

LIMITING CONDITION FOR OPERATION (Continued)

3.5.1 ACTION (Continued):

2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.

3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.

4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. For ECCS Division II, provided that ECCS Divisions I and III are OPERABLE:

1. With either LPCI subsystem "B" or "C" inoperable, restore the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 7 days.

2. With both LPCI subsystems "B" and "C" inoperable, restore at least the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

c. For ECCS Division III, provided that ECCS Divisions I and II and the RCIC system are OPERABLE:

1. With ECCS Division III inoperable, restore the inoperable division to OPERABLE status within 14 days.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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d. For ECCS Divisions I and II, provided that ECCS Division III is OPERABLE:

1. With LPCI subsystem "A" and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCI subsystem "A" or inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

2. With the LPCS system inoperable and either LPCI subsystems "B" or "C" inoperable, restore at least the inoperable LPCS system or inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

No Changes. Provided for Continuity Only.

LIMITING CONDITION FOR OPERATION (Continued)

3.5.1 ACTION (Continued):

e. For ECCS Divisions I and II, provided that ECCS Division III is OPERABLE and Divisions I and II are otherwise OPERABLE:

1. With one of the above required ADS valves inoperable, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 100 psig within the next 24 hours.

2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤ 100 psig within the next 24 hours.

f. With an ADS accumulator low pressure alarm system instrumentation channel(s) inoperable:

1. Determine the associated ADS accumulator system pressure from alternate indication and verify that ADS accumulator pressure is greater than or equal to 140 psig at least once per 12 hours,

2. Restore the inoperable ADS accumulator low pressure alarm system instrumentation channel(s) to OPERABLE status within 30 days or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status,

3. The provisions of Specification 3.0.4 are not applicable.

g. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.1 ECCS Divisions I, II, and III shall be demonstrated OPERABLE by:

a. At least once per 31 days for the LPCS, LPCI, and HPCS systems:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

4.5.1 (Continued)

2. Verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

b. Verifying that when tested pursuant to Specification 4.0.5 each:

1. LPCS pump develops a flow of at least 5010 gpm with a pump differential pressure greater than or equal to 276 psid.

2. LPCI pump develops a flow of at least 5050 gpm with a pump differential pressure greater than or equal to 113 psid.

3. HPCS pump develops a flow of at least 5010 gpm with a pump differential pressure greater than or equal to 363 psid.

c. For the LPCS, LPCI, and HPCS systems, at least once per 18 months performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

d. For the HPCS system, at least once per 18 months, verifying that the suction is automatically transferred from the RCIC storage tank to the suppression pool on a RCIC storage tank low water level signal and on a suppression pool high water level signal.

e. For the ADS by:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator low pressure alarm system.

2. At least once per 18 months, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

3. At least once per 18 months, manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig** and observing that:

   a. The control valve or bypass valve position responds accordingly, or

   b. There is a corresponding change in the measured steam flow, or

   c. The acoustic tail-pipe monitor alarms.

4. At least once per 18 months, performing a CHANNEL CALIBRATION of the accumulator low pressure alarm system and verifying an alarm setpoint of $\geq$ 140 psig on decreasing pressure.

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*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam ~~pressure is~~ conditions are adequate to perform the test.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

The recirculation flow control valves provide regulation of individual recirculation loop drive flows; which, in turn, will vary the flow rate of coolant through the reactor core over a range consistent with the rod pattern and recirculation pump speed. The recirculation flow control system consists of the electronic and hydraulic components necessary for the positioning of the two hydraulically actuated flow control valves. Solid state control logic will generate a flow control valve "motion inhibit" signal in response to any one of several hydraulic power unit or analog control circuit failure signals. The "motion inhibit" signal causes hydraulic power unit shutdown and hydraulic isolation such that the flow control valve fails "as is." This design feature insures that the flow control valves do not respond to potentially erroneous control signals.

Electronic limiters exist in the position control loop of each flow control valve to limit the flow control valve stroking rate to 10±1% per second in opening and closing directions on a control signal failure. The analysis of the recirculation flow control failures on increasing and decreasing flow are presented in Sections 15.3 and 15.4 of the USAR respectively.

The required surveillance interval is adequate to ensure that the flow control valves remain OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves (SRV) operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code. A total of 11 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. Any combination of 5 SRVs operating in the relief mode and 6 SRVs operating in the safety mode is acceptable.

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

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The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of 5 valves and lowering the opening setpoint of 2 valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

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The surveillance requirement for performing a CHANNEL CALIBRATION of the acoustic monitor(s) includes an exception to the provisions of Specification 4.0.4. This exception allows reactor steam conditions to be established which are adequate to open the SRVs without resulting in unnecessary wear on the valves and to ensure that proper reactor pressure control can be maintained while opening and reclosing the valves. Reactor steam conditions which are considered adequate to perform the test thus include the establishment of sufficient reactor pressure as well as sufficient steam flow to ensure that the steam relieved by the SRVs can be compensated by the reactor pressure control system.

No Changes. Provided for Continuity Only.

3/4.5.1 AND 3/4.5.2 ECCS - OPERATING AND SHUTDOWN

ECCS division 1 consists of the low pressure core spray system and low pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by ADS trip system "1". ECCS division 2 consists of low pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system as actuated by ADS trip system "2".

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled folowing a loss-of-coolant accident and, together with the LPCI system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The LPCS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. The LPCI system, together with the LPCS system, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

ECCS division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of 1177 psid, differential pressure between reactor vessel and HPCS suction source, to 0 psid.

3/4.5.1 and 3/4.5.2 ECCS - OPERATING AND SHUTDOWN (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 467/1400/5010 gpm at differential pressures of 1177/1147/200 psid. Initially, water from the reactor core isolation cooling (RCIC) tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the RCIC tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls seven selected safety-relief valves although the safety analysis only takes credit for six valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

Insert

3/4.5.3 SUPPRESSION POOL

The suppression pool is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression pool minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.3.1.

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The surveillance requirements for the ADS include a requirement to manually open each ADS valve. This requirement includes an exception to the provisions of Specification 4.0.4. This exception allows reactor steam conditions to be established which are adequate to open the ADS valves without resulting in unnecessary wear on the valves and to ensure that proper reactor pressure control can be maintained while opening and reclosing the valves. Reactor steam conditions which are considered adequate to perform the test thus include the establishment of sufficient reactor pressure as well as sufficient steam flow to ensure that the steam relieved by the ADS valves can be compensated by the reactor pressure control system.