ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

RELATED TO

CORRECTION OF AND

UPDATING OF MISCELLANEOUS TECHNICAL MATTERS

POWER AUTHORITY OF THE STATE OF NEW YORK
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2.1 BASES (cont'd)

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is by-passed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the Safety Limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

b. APRM Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position.

c. APRM Flux Scram Trip Setting (Run Mode)

The APRM flux scram trip in the run mode consists of a flow referenced scram setpoint and a fixed high neutron flux scram setpoint. The APRM flow referenced neutron flux signal is passed through a filtering network with a time constant which is representative of the fuel dynamics. This provides a flow referenced signal that approximates the average heat flux or thermal power that is developed in the core during transient or steady-state conditions. This prevents spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Examples of events which can result in momentary neutron flux spikes are momentary flow changes in the recirculation system flow, and small pressure disturbances during turbine stop valve and turbine control valve testing. These flux spikes represent no hazard to the fuel since they are only of a few seconds duration and less than 120% of rated thermal power.

The APRM flow referenced scram trip setting at full recirculation flow is adjustable up to 117% of

1. Refuel Mode - The reactor is in the refuel mode when the Mode Switch is in the Refuel Mode position. When the Mode Switch is in the Refuel position, the refueling interlocks are in service.

2. Run Mode - In this mode the reactor system pressure is at or above 825 psig and the Reactor Protection System is energized with APRM protection (excluding the 15 percent high flux trip) and the RBM interlocks in service.

3. Shutdown Mode - The reactor is in the shutdown mode when the Reactor Mode Switch is in the Shutdown Mode position.

    a. Hot shutdown means conditions as above with reactor coolant temperature >212° F.

    b. Cold shutdown means conditions as above with reactor coolant temperature ≤212° F and the reactor vessel vented.

4. Startup/Hot Standby - In this mode the reactor protection scram trips initiated by main steam line isolation valve closure is bypassed when reactor pressure is less than 1,005 psig, the low pressure main steam line isolation valve closure trip is bypassed, the Reactor Protection System is energized with APRM (15 percent) and IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

J. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

K. Operating - Operating means that a system or component is performing its intended functions in its required manner.

L. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

M. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

    1. All manual containment isolation valves on lines connected to the Reactor Coolant System or containment which are not required to be open during plant accident conditions are closed. These valves may be

steam line isolation valves, main steam drain valves, recirc. sample valves (Group 1), initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 18 in. above the top of the active fuel. This trip activates the remainder of the ECCS subsystems, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate ECCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10CFR100 will not be exceeded. For large breaks up to the complete circumferential break of a 24 in. recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating ECCS, it causes isolation of Groups A and B isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. See Specification 3.7 for isolation valve closure group. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperature peak at approximately 1,000°F and release of radioactivity to the environs is below 10CFR100 guidelines. Reference Section 14.6.5 FSAR.

At power levels below 20% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calories per gram drop limit. In this range, the RWM and RSCS constrain the control rod sequence and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviance from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences which limit the maximal reactivity worth of control rods , in the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator or other qualified technical plant employee whose qualifications have been reviewed by the NPC can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural control to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20%, these devices force adherence to acceptable rod patterns. Above 20% of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20% of rated power are imposed by power distribution requirements as defined in Section 3.3.3.5 of these Technical Specifications. Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of ± 2% of full power, the nominal instrument setting is 22% of rated power. Power level for automatic cutout of the RWM function is sensed by steam flow and is set manually at 30% of rated power to be consistent with the RSCS setting.

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 20% rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

The RSCS can be functionally tested prior to control rod withdrawal for reactor startup. By selecting, for example, A_{12} and attempting to withdraw, by one notch, a rod or all rods in each other group, it can be determined that the A_{12} group is exclusive. By bypassing to full-out all A_{12} rods, selecting A_{34} and attempting to withdraw, by one notch, a rod or all rods in group B, the A_{34} group is determined exclusive. The same procedure can be repeated for the B groups. After 50% of the control

H. Jet Pump Flow Mismatch

1. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50 percent of its rated speed.

2. The reactor shall not be operated for a total period in excess of 24 hours with one or more recirculation loops out of service.

1. The two recirculation loops have a flow imbalance of 15 percent or more when the pumps are operated at the same speed.

2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10 percent.

3. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the average of all jet pump differential pressures by more than 10 percent.

would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle-riser system failure.

H. Jet Pump Flow Mismatch

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50 percent of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

I. Hydraulic Shock Suppressors

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.1.4 prohibits startup with inoperable snubbers.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less than 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment.

2. From and after the date that one of the Core Spray Systems is made or found inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the system is made operable earlier, provided that during the 7 days all active components of the other Core Spray System and the LPCI System and the emergency diesel generators shall be operable.

3. The LPCI mode of the RHR System shall be operable whenever irradiated fuel is in the reactor and prior to reactor startup from a cold condition, except as specified below.

   a. From the time that one of the RHR pumps is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the pump is made operable earlier provided that during such 7 days the remaining active components of the LPCI, containment spray mode, all active components of both Core Spray Systems, and the emergency diesel generators are operable.

2. When it is determined that one Core Spray System is inoperable, the operable Core Spray System, the LPCI System, and the emergency diesel generators shall be demonstrated to be operable immediately. The remaining Core Spray System shall be demonstrated to be operable daily thereafter.

3. LPCI System testing shall be as specified in 4.5.A.1.a, b, c, d, f and g except that each RHR pump shall deliver at least 9,900 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.

   a. When it is determined that one of the RHR pumps is inoperable, the remaining active components of the LPCI, containment spray subsystem, both Core Spray Systems, and the emergency diesel generators required for operation shall be demonstrated to be operable immediately, and the remaining RHR pumps shall be demonstrated to be operable daily thereafter.