UNITED STATED OF AMERICA
BEFORE THE NUCLEAR REGULATORY COMMISSION

AFFIDAVIT OF BEN L. RIDINGS

I, Ben L. Ridings do make oath and say:

1. My name is Ben L. Ridings. I am a technical consultant for commercial nuclear power plants. Over a span of some fifteen years, while working at some twenty four sites, I have specialized in reviewing of licensing agreement (FSAR, Technical Specifications, Federal Codes and Regulations, ASME Codes, etc.), establishing administrative controls to meet these requirements and test programs to ensure compliance at all times. My test programs and administrative controls established while under contract to arious utilities are still in use today at many facilities.

2. I have re iewed all of the relevant pulicly available correspondence between the Nuclear Regulatory Commission and Niagra Mohawk during the relvant time span. I am familar with NRC regulations and regulatory guidance governing High Pressure Core Injection.

3. The factual statement made in the attached Petition for Emergency Action and Request for public Hearing are true and correct to the best of my knowlege and belief.

Ben L Ridings

Ben L. Ridings

Subscribed and sworn to before me this 28th day of Oct, 1992.

Mary Duke

My commision expires: 5-21-95

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Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33—Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34—Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 36—Inspection of emergency core cooling system. The emergency core

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cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38—Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40—Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41—Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Criterion 42—Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43—Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44—Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

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LIMITING CONDITION FOR OPERATION

3.1.8 HIGH PRESSURE COOLANT INJECTION

Applicability:

Applies to the operational status of the high pressure coolant injection system.

Objective:

To assure the capability of the high pressure coolant injection system to cool reactor fuel in the event of a loss-of-coolant accident.

Specification: 1,2H**

a. During the power operating condition whenever the reactor coolant pressure is greater than 110 psig and the reactor coolant temperature greater than saturation temperature, the high pressure coolant injection system shall be operable except as specified in Specification "b" below.

b. If a redundant component of the high pressure coolant injection system becomes inoperable the high pressure coolant injection shall be considered operable provided that the component is returned to an operable condition within 15 days and the additional surveillance required is performed.

SURVEILLANCE REQUIREMENT

4.1.8 HIGH PRESSURE COOLANT INJECTION

Applicability:

Applies to the periodic testing requirements for the high pressure coolant injection system.

Objective:

To verify the operability of the high pressure coolant injection system.

Specification:

The high pressure coolant injection surveillance shall be performed as indicated below:

a. At least once per operating cycle -

Automatic start-up of the high pressure coolant injection system shall be demonstrated.

b. At least once per quarter -

Pump operability shall be determined.

LIMITING CONDITION FOR OPERATION

c. If Specification "a" and "b" are not met, a normal orderly shutdown shall be initiated within one hour and reactor coolant pressure and temperature shall be reduced to less than 110 psig and saturation temperature within 24 hours.

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SURVEILLANCE REQUIREMENT

c. Surveillance with Inoperable Component

When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

BASES FOR 3.1.8 AND 4.1.8 HIGH PRESSURE COOLANT INJECTION

e High Pressure Coolant Injection System (HPCI) is provided to ensure adequate core cooling in the unlikely event of all reactor coolant line break. The HPCI System is required for line breaks which exceed the capability of the ntrol Rod Drive pumps and which are not large enough to allow fast enough depressurization for core spray to be fective.

e set of high pressure coolant injection pumps consists of a condensate pump, a feedwater booster pump and a motor iven feedwater pump. One set of pumps is capable of delivering 3,800 gpm to the reactor vessel at reactor essure. The performance capability of HPCI alone and in conjunction with other systems to provide adequate core oling for a spectrum of line breaks is discussed in the Fifth Supplement of the FSAR.

determining the operability of the HPCI System, the required performance capability of various components shall be nsidered.

The HPCI System shall be capable of meeting its pump head versus flow curve.

The motor driven feedwater pump shall be capable of automatic initiation upon receipt of either an automatic turbine trip signal or reactor low-water-level signal.

The Condenser hotwell level shall not be less than 57 inches (75,000 gallons).

The Condensate storage tanks inventory shall not be less than 105,000 gallons.

The motor-driven feedwater pump will automatically trip if reactor high water level is sustained for ten seconds and the associated pump downstream flow control valve and low flow control valve are not closed.

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ring reactor start-up, operation and shutdown, the condensate and feedwater booster pumps are in operation. At eactor pressures up to 450 psig, these pumps are capable of supplying the required 3,800 gpm. Above 450 psig a otor-driven-feedwater pump is necessary to provide the required flow rate.

he capability of the condensate, feedwa' · booster and motor driven feedwater pumps will be demonstrated by their peration as part of the feedwater supply during normal station operation. Stand-by pumps will be placed in service t least quarterly to supply feedwater during station operation. An automatic system initiation test will be erformed at least once per operating cycle. This will involve automatic starting of the motor driven feedwater pumps nd flow to the reactor vessel.

I. HIGH-PRESSURE COOLANT INJECTION

1.0 Design Bases

The high-pressure coolant injection (HPCI) system is an operating mode of the feedwater system available in the event of a small reactor coolant line break which exceeds the capability of the control rod drive pumps (0.003 ft^2). HPCI along with one emergency cooling system has the capability of keeping the swollen reactor coolant level above the top of active fuel for small reactor coolant boundary breaks up to 0.07 ft^2 for at least 1000 seconds. The HPCI system with one of the two emergency cooling systems and two core spray systems, will provide core cooling for the complete spectrum of break sizes up to the maximum design basis recirculation discharge line break (5.446 ft^2). Its primary purpose is to: Rev. 7

a. provide adequate cooling of the reactor core under abnormal and accident conditions.

b. remove the heat from radioactive decay and residual heat from the reactor core at such a rate that fuel clad melting would be prevented.

c. provide for continuity of core cooling over the complete range of postulated break sizes in the primary system process barrier.

HPCI is not an engineered safeguards system and is not considered in any Loss of Coolant Accident Analyses. It is discussed in this section because of its capability to provide makeup water at reactor operating pressure. Rev. 7

2.0 System Design

The HPCI system utilizes the two condensate storage tanks, the main condenser hotwell, two condensate pumps, condensate demineralizers, two feedwater booster pumps, feedwater heaters, two motor-driven feedwater pumps, an integrated control system and all associated piping and valves. The system is capable of delivering 7600 gpm into the reactor vessel at reactor pressure when using two trains of feedwater pumps. The condensate and feedwater booster pumps are capable of supplying the required 3,800 gpm at approximately reactor pressures up to 270 psig. Above 270 psig a motor-driven feedwater pump is necessary to provide the required flow rate. Rev. 7

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The feedwater system pumps have recirculation lines with air operated flow control valves to prevent the pumps from operating against a closed system. In the event of loss of air pressure, these valves open recycling part of the HPCI flow to the hotwell. HPCI flow would be reduced to approximately 3,000 gpm at a reactor pressure of 1,150 psig and 3,800 gpm at a reactor pressure of 940 psig. Rev. 7

Condensate inventory is maintained at an available minimum volume of 180,000 gallons.

3.0 Design Evaluation

During a loss-of-coolant accident within the drywell, high drywell pressure due to a line break will cause a reactor scram. This automatic scram will cause a turbine trip after a five-second delay. In order to prevent cladding temperatures from exceeding their maximum limit for the entire spectrum of breaks, the 3800 gpm (from one train of HPCI/feedwater pumps) would have to be available immediately. Feedwater flow would be available for considerable time from the shaft-driven feedwater pump. The shaft-driven feedwater pump would coast down while the electric motor-driven condensate pumps and feedwater booster pumps would continue to operate. The coast down time to reach 3,800 gpm delivery to the core is approximately 3.2 minutes (Figure VII-17), since both the condensate and feedwater booster pumps will continue to operate on off-site power. Rev. 7

The turbine trip will signal the motor-driven feedwater pump to start. The signal will be simultaneous with the start of the shaft pump coast down. The motor-driven feedwater pump will be up to speed and capable of supplying 3,800 gpm in about ten seconds. As a backup, low reactor water level will also signal the motor-driven pump to start. The initiation signal transfers control from the normal feedwater to the HPCI instrumentation and controller which has been continuously tracking the normal feedwater control signal. Thus there will be a continuous supply of feedwater to the reactor.

The HPCI single element control system will attempt to maintain reactor vessel water level at 65 inches or 72 inches (depending upon which pump, 11 or 12 respectively, is in service) with a maximum feedwater flow limit of 3800 gpm.

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A sustained high reactor water level reactor protection system signal coincident with an open feedwater flow control valve will selectively trip the associated feedwater pump. The clutch of the shaft-driven pump will also be disengaged immediately upon high reactor water level.

Should the reactor water level reach the low level scram setpoint the motor driven pump that tripped on high reactor water level will restart. Necessary feedwater pump recirculation is provided to allow for continued pump operation with the flow control valve closed.

As feedwater is pumped out of the condenser hotwell, through the selected equipment of the condensate and feedwater systems and into the reactor, the condenser hotwell level will fall. Since condensed steam from the turbine no longer replenishes the condenser hotwell, condensate will be transferred from the condensate storage tanks to the hotwell for makeup.

The feedwater system pumps operate on 4160 v. When the plant is in operation, the power is supplied from the main generator through the station service transformer when the generator is on-line and connected to the grid. When the main generator is off-line, the feedwater pumps are supplied with normal off-site power from the 115 KV system through the reserve transformers. If a HPCI initiation signal should occur, all HPCI/feedwater system pumps would start immediately with two feedwater pump trains available for HPCI injection using the single element feedwater control system for reactor vessel level control. If a major power disturbance were to occur that resulted in loss of the 115 KV power supply to the Nine Mile Point 115 KV bus, power would be restored from a generator located at the Bennetts Bridge Hydro Station. This generator would have the capacity of supplying approximately 6,000 KVA which is sufficient to operate one train of HPCI/feedwater system pumps. If HPCI initiation were to occur, the preferred feedwater train pumps (feedwater pump 12, feedwater booster pump 13, condensate pump 13) would start. The non-preferred train pumps would be locked out on loss of off-site power and not start until the operator manually reset the lock out. If a preferred train pump had been locked out prior to the loss of off-site power, it would remain locked out and the non-preferred train backup pump would automatically start on HPCI initiation. If both the preferred and backup pumps are running, the preferred pump would remain in service and the backup pump will trip. The

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use of a Bennetts Bridge hydro generator, while not equivalent to an on-site emergency power source, provides a highly reliable alternate off-site power supply for the HPCI function of the feedwater system. Rev. 7

4.0 Tests and Inspections

Tests and inspections of the various components are described in Section XI - Steam to Power Conversion.

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COMPONENT ID #	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	INDENTIFIED IN TS	TS SIGNAL	TS ST	PBID PRINTS ID RPS LOGIC	NOTE
05-03	Y	CLOSE	5	N	—	—	*	9
05-02	Y	CLOSE	5	N	—	—	*	9
39-12	Y	CLOSE	5	N	—	—	Y	9
39-11	Y	CLOSE	5	N	—	—	Y	9
39-13	Y	CLOSE	5	N	—	—	Y	9
39-14	Y	CLOSE	5	N	—	—	Y	9
40-12	Y	OPEN	—	N	—	—	Y	1
40-01	Y	OPEN	—	N	—	—	Y	1
40-09	Y	OPEN	—	N	—	—	Y	1
40-11	Y	OPEN	—	N	—	—	Y	1
40-10	Y	OPEN	—	N	—	—	Y	1
40-02	Y	OPEN	—	N	—	—	Y	1
58.1-01	N	—	—	Y	—	60	Y	10
80-118	N	—	—	N	—	—	—	2
30-31	N	—	—	N	—	—	HPCI	13
30-32	N	—	—	N	—	—	HPCI	13
201.1-14	N	—	—	N	—	—	Y	11
201.1-16	N	—	—	N	—	—	Y	11
201.1-09	N	—	—	N	—	—	Y	11
201.1-11	N	—	—	N	—	—	Y	11
63-04	Y	CLOSE	30	N	—	—	Y	12
63-05	Y	CLOSE	30	N	—	—	Y	12
80-15	Y	—	60	Y	OPEN	60	Y	3/17
80-16	Y	—	60	Y	OPEN	60	Y	3/17
80-35	Y	—	60	Y	OPEN	60	Y	3/17
80-36	Y	—	60	Y	OPEN	60	Y	3/17

* - see note 9

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COMPONENT # F	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	IDENTIFIED IN TS	TS SIGNAL	TS ST	P&ID PRINTS ID RPS LOGIC	NOTES
201.2-109	Y	CLOSE	60	N	—	—	Y	11
201.2-112	Y	CLOSE	60	N	—	—	Y	11
44.2-15	Y	CLOSE	18	Y	CLOSE	10	Y	4
44.2-16	Y	CLOSE	18	Y	CLOSE	10	Y	4
40-30	Y	CLOSE	30	Y	CLOSE	30	Y	5
40-31	Y	CLOSE	30	Y	CLOSE	30	Y	5
40-32	Y	CLOSE	30	Y	CLOSE	30	Y	5
40-33	Y	CLOSE	30	Y	CLOSE	50	Y	5
122-03	Y	CLOSE	30	N	—	—	Y	12
110-127	Y	CLOSE	20	N	—	—	Y	9
110-128	Y	CLOSE	20	N	—	—	Y	9
202-07	Y	CLOSE	60	N	—	—	Y	12
202-08	Y	CLOSE	60	N	—	—	Y	12
02-35	Y	CLOSE	60	N	—	—	Y	12
.02-36	Y	CLOSE	60	N	—	—	Y	12
30-114	Y	CLOSE	90/70	N	—	—	Y	6
30-115	Y	CLOSE	90/70	N	—	—	Y	6
201.7-08	N	—	—	—	—	—	Y	7
201.7-09	N	—	—	—	—	—	Y	7
201.2-25	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-26	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-27	Y	CLOSE	60	Y	CLOSE	60	Y	8
01.2-28	Y	CLOSE	60	Y	CLOSE	60	Y	8
01.2-29	Y	CLOSE	60	Y	CLOSE	60	Y	8
0.2-30	Y	CLOSE	60	Y	CLOSE	60	Y	8
01.2-23	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-24	Y	CLOSE	60	Y	CLOSE	60	Y	8

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COMPONENT I.D.	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	IDENTIFIED IN TS	TS SIGNAL	TS ST	P&ID PRINTS ID RPS LOGIC	NOTES
201.7-01	Y	CLOSE	60	N	—	—	Y	11
01.7-02	Y	CLOSE	60	N	—	—	Y	11
01.7-03	Y	CLOSE	60	N	—	—	Y	11
01.7-04	Y	CLOSE	60	N	—	—	Y	11
01.7-10	Y	CLOSE	60	N	—	—	Y	11
01.7-11	Y	CLOSE	60	N	—	—	Y	11
01.2-110	Y	CLOSE	60	N	—	—	Y	11
01.2-111	Y	CLOSE	60	N	—	—	Y	11
34-01	Y	—	30	Y	—	30	—	10
29-51	N	—	—	N	—	—	HPCI	13
29-51	N	—	—	N	—	—	HPCI	13
1P-1	Y	CLOSE	60	N	—	—	**	14
1P-2	Y	CLOSE	60	N	—	—	**	14
1P-3	Y	CLOSE	60	N	—	—	**	14
1P-4	Y	CLOSE	60	N	—	—	**	14
0-94	Y	—	30	Y	—	30	N	15/16
0-92	Y	—	30	Y	—	30	N	16
70-93	Y	—	—	Y	—	—	N	16
70-94	Y	—	—	Y	—	—	N	16
80-17	Y	—	—	Y	—	—	N	17
80-18	Y	—	—	Y	—	—	N	17
80-37	Y	—	—	Y	—	—	N	17
80-38	Y	—	—	Y	—	—	N	17
80-65	Y	—	—	Y	—	—	N	17
-66	Y	—	—	Y	—	—	N	17
80-67	Y	—	—	Y	—	—	N	17
80-68	Y	—	—	Y	—	—	N	17

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(4)

COMPONENT #	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	IDENTIFIED IN TS	TS SIGNAL	TS ST	P&ID PRINTS ID RPS LOGIC	NOTES
80-01	Y	–	70	Y	–	70	N	17
80-02	Y	–	70	Y	–	70	N	17
80-21	Y	–	70	Y	–	70	N	17
80-22	Y	–	70	Y	–	70	N	17
81-01	Y	–	70	Y	–	70	N	18
81-02	Y	–	70	Y	–	70	N	18
81-21	Y	–	70	Y	–	70	N	18
81-22	Y	–	70	Y	–	70	N	18

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NOTES:

(1) FSAR section VII requires these valves to go open within 20 sec Hi Drywell or Lo Lo Rx Level RPS signal and this time fails to appear in either TS Table 3.3.4 or FSAR Table VI-3a. Also, these valves are 10CFR50 Appendix A Criterion 55 valves and are not being tested accordingly.

(2) Containment Spray Test Line currently does not recieve RPS signal to go closed. The effectiveness of one containment spray pump is lost until operator response manually closes valve should the accident occur during testing of containment spray pumps. Also, this is a criterion 56 valve and is not being tested accordingly and should appear in TS 3.2.7 and FSAR Table VI-3b.

(3) FSAR Table VI-3b shows these valves recieve NO RPS signal. TS Table 3.3.4 shows these valves recieve signal to open. P&ID C18012C shows RPS logic to these valves. Also, these are Criterion 56 valves valves and are not being tested accordingly.

(4) FSAR Table VI-3a shows a close stroke time of 18 seconds while TS Table 3.2.7 shows 10 second closure. Even though this is more conservative, the discrepancy came about as an error because components are not individually listed in tables.

Notes
Cont.

(5) FSAR Table 3a shows RPS logic to close with core spray actuation while TS Table 3.27 does not.

(6) FSAR Table 3b shows these valves with a 70 second and 90 second stroke time. These valves should also appear on TS Table 3.3.4.

(7) P&ID 18014C sht 2 shows these valves recieve an RPS signal however. FSAR Table VI-3b and TS Table 3.3.4 fail to include these penetrations and stroke times.

(8) These value are Criterion 56 valves which appear in FSAR Table VI-3b. These valves may or may not (see note 12) appear in TS Table 3.3.4. TS as written, it is impossible to distinguish however these values are identified in surveillance test (N1-ST-Q5) as TS acceptance criteria.

(9) FSAR Table VI-3a shows RPS logic to close however TS Table 3.2.7 does not identify these valves. Also, valves (*) appear on P&ID C18006C with no RPS logic while they are identified with RPS logic on P&ID C18017C.

(10) These valves are deactivated and the TS and appropriate FSAR sections should be revised to reflect this change.

NOTES CONT.

(11) These valves are identified on P&ID C18014C sht 1 as recieving RPS logic yet do not appear in FSAR Table VI-3b or TS Table 3.3.4.

(12) FSAR Table VI-3b show RPS logic to close however TS Table 3.3.4 does not identify these valves. Effects Surveillance Program and procedure revision,

(13) P&ID C18005C sht 1 show HPCI logic to close yet are not identied in TS or FSAR. Also not identified in IST Program.

(14) FSAR Table VI-3b show RPS logic to close however TS Table 3.3.4 does not identify these valves. Also, tested IAW N1-ST-Q5, current procedure 5 sec TS acceptance criteria that does not exist. Also these valves do not appear on drawings C18014C as identified in IST Plan.

(15) Currently tested IAW N1-ST-Q7 with IST acceptance criteria of 60 sec. No FSAR or TS stroke times identified.

(16) FSAR VI-3c identifies these valves as criterion 57 valves. TS Table 3.3.4 identifies these valves as both criterion 56 and 57 valves. This is physically impossible. Secondly, these valves are not tested to either criterion.

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NOTES
CONT

(18) FSAR Table VI-3b and TS Table 3.3.4 identify these values as criterion 56 valves however are not being tested accordingly.