**Murray Selman**
Vice President

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Broadway & Bleakley Avenue
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Telephone (914) 737-8116

September 12, 1986

Re: Indian Point Unit No. 2
Docket No. 50-247

Mr. Hugh L Thompson, Jr., Director
Division of PWR Licensing - A
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Thompson:

This is in response to Ms. Slosson's letter of June 30, 1986 which forwarded a preliminary technical evaluation of our NUREG-0737, Supplement 1 (Generic Letter 82-33) review of post accident monitoring instrumentation at Indian Point Unit 2 (IP-2) submitted on August 30, 1985. In that submittal, we evaluated the degree of compliance of IP-2's design basis with the guidance contained in Regulatory Guide 1.97, Revision 2, as required by Generic Letter 82-33. Recognizing that Regulatory Guide 1.97, Rev. 2 is a guidance document, we evaluated exceptions based on plant-specific features of IP-2, and the functional requirements of all the initiatives of NUREG-0737, Supplement 1. Our evaluation showed that, in some cases, the guidance in Regulatory Guide 1.97, Rev. 2 was applicable and commitments were accordingly made to install the necessary instrumentation to meet NUREG-0737, Supplement 1; in other cases, justifications and alternatives were presented for the exceptions since installing such additional instruments was not necessary to assist plant operators in diagnosing and mitigating the consequences of design basis accidents.

We have determined that our August 30, 1985 submittal meets NUREG-0737, Supplement 1 requirements in an integrated fashion with the other Supplement 1 initiatives, and that there is no need to upgrade IP-2 to specifically meet the guidance of Regulatory Guide 1.97, Rev. 2 in every instance. In your June 30, 1986 evaluation some of those justifications and alternatives were identified as being unacceptable, and instrumentation upgrading was requested; the reasons for the item being unacceptable were, however, not explicit. Absent such reasoning, we are led to conclude that Regulatory Guide 1.97, Rev. 2 alone was used as the acceptance criteria and not an integrated implementation of NUREG-0737, Supplement 1. As noted in Supplement 1, guidance documents such as Regulatory Guide 1.97, Rev. 2 shall not be misconstrued as requirements

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or as inflexible criteria. Therefore, we request that our response to your June 30, 1986 evaluation as contained in the Attachments to this letter should be reviewed as part of a larger integrated program which we are implementing to achieve an improved man-machine interface intended to assist a plant operator in diagnosing and coping with transients and accidents as required by Generic Letter 82-33.

Attachment 1 contains our responses to the open items in the above-noted preliminary technical evaluation. Since the time of our August 30, 1985 submittal we have been incorporating the Supplement 1 retrofit projects into IP-2. This has caused us to frequently re-assess our engineering evaluations and background analyses which led to our commitments to assure proper integration of the Supplement 1 initiatives. Additionally, since the upgraded emergency operating procedures have been in place for nearly one year, we have gained valuable feedback which will better assist us in phasing these retrofit projects into the control room. As a result of these assessments, it has become apparent to us that certain portions of our August 30, 1985 submittal need to be revised. These revisions are contained in Attachment 2 and are denoted by bars in the margin. Where these revisions impact an open item in the preliminary technical evaluation report, it is discussed in the response in Attachment 1.

Should you or your staff have any questions on this matter, please contact me.

Very truly yours,

Murray Selman

Murray Selman
Vice President

cc: Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

Attachment 1

Response To Preliminary Technical
Evaluation Report

September, 1986

This Attachment presents an item by item response to the open items identified in the preliminary technical evaluation report transmitted by M. Slosson's letter of June 30, 1986.

1) Neutron Flux (item 3.3.1):

This is a category 3 variable which we are proposing not to upgrade. We have reviewed the transients and accidents in which this variable is relied upon and determined that the present system is adequate since this is not a key variable for determining either the state of the core during worst case accidents or if an accident has been successfully mitigated. The justification presented in our August 30, 1985 submittal was mainly aimed at showing that, for worst case design basis loss of coolant accidents (LOCAs) inside containment we would not have to rely on neutron flux indication. This was based on evaluating plant conditions given that neutron flux was either not reading or was reading erroneously. In each case, successful detection of criticality and accident mitigation can be achieved by utilizing alternative plant equipment, which is either qualified or committed for upgrade. There are many other design basis events which were not discussed in our August 30, 1985 submittal where the core heatup, resulting from an unwanted post accident criticality, would be detected by both the neutron flux readings and core exit temperature. However, for these types of events, the containment environment would not become harsh, and thus the neutron flux instrumentation would be functional, and the current instrumentation is therefore acceptable. In the design basis cases, where the containment environment may become harsh, neutron flux is only a backup variable. The basis of this evaluation is presented below.

a) Large and Intermediate Break LOCA:

The fundamental characteristic of the large break LOCA is a rapid depressurization of the reactor coolant system (RCS). Since the RCS blowdown area is less for smaller size breaks, the intermediate break (greater than 2" diameter) will depressurize the RCS more slowly than the large break. RCS breaks greater than 2 inches result in the classic LOCA scenario that is not dependent on secondary system cooling and requires injection of the inventory from the Refueling Water Storage Tank (RWST). The depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. A safety injection actuation signal is generated when the appropriate low-low pressure setpoint is reached. For the large break LOCA these signals occur essentially simultaneously and instantaneously with the break. For breaks greater than 2" diameter these setpoints are still reached very quickly since they are set only about 350 psi below normal operating pressure. These protective countermeasures limit the consequences of the LOCA

in two ways. First, reactor trip and borated water injection complement void formation by causing a rapid power reduction to fission product decay heat levels. Second, the injection of borated water provides for heat transfer from the core, prevents excessive fuel clad temperature, and maintains subcriticality.

The blowdown of RCS fluid into the containment causes pressure, humidity and temperature levels in containment to rise and, when containment pressure reaches the high-high containment pressure setpoint, chemical spray is injected into the containment atmosphere. During the LOCA, the blowdown and spray fluids mix with the containment air and result in a harsh environment in which exposed equipment, relied on to detect and mitigate the event, must function. However, in comparison to the rapid RCS depressurization (which causes reactor trip), the containment environment becomes harsh somewhat slower and reaches a threshold where it is mitigated by containment safeguards systems. For the intermediate break LOCA (greater than 2" diameter), the RCS depressurization is slower, the containment environment becomes harsh more slowly, and depending on the break size, may not even reach the containment spray setpoint. The reactor trip signal and resultant control rod insertion occur within a very short period from event initiation and is indicated in the control room by several diverse means before harsh environment occurs. Once the reactor is tripped, subcriticality is maintained by the continued injection of borated water as discussed above. Since neutron flux is not the only indication of reactor trip and it is not solely relied on for reactor trip verification, it is acceptable as Category 3 to perform its intended function early in a large and intermediate break LOCA scenario. If an operator cannot verify that the reactor has tripped when it is required to be tripped, he is directed by the Emergency Operating Procedures (EOP) to manually trip the reactor. If reactor trip still can not be verified, he is directed to commence emergency boration of the RCS, which for the large, and intermediate break LOCA, is already in progress due to safety injection. The operator would then proceed, as directed by the EOPs, to continue monitoring critical safety functions such as core cooling to assure that excessive fuel temperatures are not approached. Thus, for a large or intermediate break LOCA, since the reactor is reflooded with borated water from the RWST and, the contents of this tank and the ECCS accumulators are emptied prior to switchover to recirculation, there is, by design of the ECCS, a sufficient supply of borated water to maintain the reactor in a subcritical condition. Once this accident is successfully mitigated and recovery is proceeding on long term recirculation of borated cooling

water, periodic sampling to determine adequate boron concentration to maintain subcriticality is sufficient action to assure that any approach to criticality is detected, without the need for neutron flux indication. As we described in our August 30, 1985 submittal, in the highly unlikely event that the reactor should become critical in the long term recovery following mitigation of a large or intermediate break LOCA in the period between boron concentration sampling activities, the core heatup (which would be starting from essentially a cold condition) resulting from the return-to-power will be detected by the continued monitoring of the core cooling critical safety function which relies on core exit temperature and which we have already committed to upgrade. As such, for the large and intermediate break LOCA, which yields a harsh containment environment, neutron flux is not relied on to detect an approach to criticality nor to determine that the accident has been successfully mitigated. Installation of category 1 instrumentation will not provide any significant safety benefit.

b) Small Break LOCA:

Unlike the large or intermediate break LOCA, small breaks less than about two inches in diameter will result in a slower RCS depressurization, less energy input to the containment and a less severe fuel temperature transient. In fact, for breaks less than about two inches in diameter, the resultant containment temperature is not predicted to exceed the steam saturation pressure corresponding to the adverse containment criteria of the EOPs (4.0 psig). Thus, since the adverse containment criteria are not going to be exceeded, a harsh environment will not develop as with the larger breaks and Category 3 instrumentation is adequate to perform its intended function throughout the scenario. The heat removal capacity of the containment fan cooler units is more than adequate to quickly return containment conditions to normal and maintain such thereafter. Below the EOP adverse containment criterion (4.0 psig) the sensor post-accident deviation errors are relatively insignificant and one would not expect the neutron flux indication to behave abnormally even if it were to be used. In any case, the design basis of IP-2 has been to allow the RCS to depressurize with borated makeup flow provided by the ECCS and long term recirculation core cooling when the RWST is emptied. Thus, the previous discussion for large and intermediate break sizes can also be applied to smaller breaks to show neutron flux is neither relied on to detect an approach to criticality nor to determine that the accident has been successfully mitigated. Hence, installation of category 1 instrumentation will not provide any significant safety benefit.

c) Main Steam Line Break:

Breaks outside containment will not adversely affect the neutron flux detectors and thus Category 3 instrumentation will perform its intended function for these events. The main steam line break (MSLB) must occur inside containment in order for a harsh environment to be created.

The fundamental characteristic of a MSLB is a rapid cooldown and depressurization of the intact RCS due to the uncontrolled heat removal via the high blowdown steam flow out the break. The steam generator (S/G) blowdown causes a rapid pressure decrease in the faulted S/G, which when coupled with the high steam flow, initiates reactor trip, safety injection and MSIV closure. The rapid RCS cooldown causes a positive reactivity insertion due to the negative moderator temperature coefficient and quickly causes a return to criticality. The reactivity transient is mitigated by the automatic injection of borated water from the RWST via the safety injection system and, within minutes, the reactor is subcritical again. By that time, the S/G blowdown into containment is nearly complete. RCS temperature and pressure stabilizes and temperature is controlled by the three remaining intact S/Gs. The safety injection termination criteria are subsequently met when pressurizer level returns, RCS pressure is stable or increasing, and adequate subcooling margin exists. Auxiliary feedwater is throttled to maintain S/G level in the three intact S/Gs and to control RCS temperature. The RCS is basically returned to a stable thermodynamic condition below hot shutdown temperature and pressure conditions.

The action of safety injection during this accident is to emergency-borate the RCS and accommodate RCS inventory shrinkage due to rapid cooldown. During the return to criticality, the RCS heat is adequately removed by the blowdown of steam from the faulted S/G. Since it is a design basis condition to return to criticality for the MSLB, automatic core protection is provided in the form of emergency boration via the safety injection system and thus, no operator action is required to mitigate this reactivity transient. Therefore, instrumentation to detect this early criticality is not required. It should be noted that the core reactivity transient would be terminated even without the addition of boric acid when the affected S/G boils dry.

Once the S/G blowdown phase is over and the RCS temperature is stablized, the core average temperature again follows the core heat flux in the intact RCS which in turn would follow core criticality were it to occur. Core heat increases are indicative of reactivity increases that would accompany any approach or subsequent return to criticality. In the

unlikely and unpredicted event that a return-to-critical were to occur, the core average temperature would increase and, with either forced or natural circulation in the RCS, be detected by the hot leg temperature (wide range) indication which is already a category 1 variable. Also, the core exit temperature would increase and be detected by the core exit thermocouple instrumentation which we have already committed to upgrade. The mitigating actions would be the same as in the LOCA cases discussed above. RCS boron sampling would be used to verify adequate boron concentrations. Therefore, installation of category 1 neutron flux instrumentation will not provide any significant safety benefit.

d) Anticipated Transient Without Scram (ATWS):

ATWS is not a design basis event. However, since the initiating event is not a primary or secondary coolant system break, detection and mitigation of an ATWS condition does not initially involve a harsh environment inside containment. The current instrumentation would initially be functional and, with other instruments, would assist the operator in diagnosing the event and initiating emergency boration. In the worst case ATWS, assuming failure of emergency boration systems (a highly improbable event), the pressurizer safety and relief valves will lift until Doppler effect shuts down the reactor. The lifting of the valves essentially causes a LOCA and the mitigating actions would be the same as in the LOCA cases discussed above once the RCS depressurizes below the safety injection pump shutoff head. Thus, the currently installed category 3 neutron flux instrumentation is adequate for its intended function. Installation of category 1 instrumentation will not provide any significant safety benefit.

e) Other Design Basis Transients:

Other design basis transients analyzed in the FSAR, either do not involve a primary or secondary coolant system break or, if they do, the break is bounded by the cases described above, or the break is outside of containment. Therefore, the currently installed category 3 neutron flux instrumentation is adequate for its intended function. Installation of category 1 instrumentation will not provide any significant safety benefit.

2) Degrees Of Subcooling (3.3.4):

The subcooling margin monitor (SMM) meets category 1 requirements except for the seismic qualification of the monitor and computer. When we addressed this instrumentation in our August 30, 1985 submittal we classifed the SMM as category 2 instead of category 1 with an exception, since the SMM met category 2 requirements.

The SMM is a single readout. A second readout will be provided by the Safety Assessment System/Safety Parameter Display System (SAS/SPDS) top level display and would be used by an operator in much the same way as the SMM. The redundancy requirements of

category 1 will then be met. The inputs signals to the SMM and SAS/SPDS for subcooling are already qualified and redundant and are appropriately isolated from their individual panel displays. The SMM and SAS/SPDS displays and computers are located in non-harsh environments. Thus, with the exception of the seismic qualification, this instrumentation is category 1. The backup system for obtaining subcooling margin information utilizes RCS wide range pressure and temperature instrumentation directly without reliance on the SMM or SAS/SPDS displays or computers. The wide range pressure and temperature instrumentation is category 1. Thus, should the SMM or SAS/SPDS be unavailable during a design basis seismic event, seismically qualified instrumentation is available to determine subcooling, as necessary. Therefore, installation of seismically qualified monitors and computers will not provide any significant safety benefit. The Table of Variables contained in Attachment 2 to this submittal has been revised accordingly.

3) Residual Heat Removal Heat Exchanger Outlet Temperature (3.3.9):

RHR heat exchanger outlet temperature will be upgraded to meet category 2 environmental qualification requirements by startup from the Cycle 9/10 refueling outage (currently scheduled for about second quarter 1989). The Table of Variables has been revised as shown in Attachment 2. This commitment also modifies the justification provided in Note 24 the Table of Variables.

4) Accumulator Tank Level and Pressure (3.3.10) and Accumumlator Isolation Valve Position:

In analyzing these variables, we conservatively assumed that if a variable was to be classified as category 1 or 2 by Table 2 of Reg. Guide 1.97, Rev. 2, it had to be environmentally qualified to the worst case equipment qualification envelope used for safety grade equipment inside containment. We then classified these IP-2 variables under that conservative assumption and tried to justify that these should be category 3. The June 30, 1986 technical evaluation report indicated for these variables that they should be qualified in accordance with 10 CFR 50.49. Since the time dependent pressure and temperature envelope for this equipment when it is relied on to function, is no more severe than that for normal operation, then it is does not fall within the scope of 10 CFR 50.49 since it functions under normal operating conditions and is not relied on to function under design basis accident conditions yielding a harsh operating environment. The IP-2 EOPs do not rely on these variables to cope with accident conditions. Thus, this instrumentation does not fall under the scope of post-accident monitoring variables required by Supplement 1. Accordingly, the Table of Variables contained in our August 30, 1985 submittal has been revised (see Attachment 2) to indicate that these variables are not applicable.

Neither of these are key variables to determine the status of the tanks in an accident condition. The accumulators are an automatic passive part of the ECCS in the IP-2 design. During the RCS blowdown phase of a large break LOCA, the lower reactor vessel plenum quickly empties. Since this is an expected condition,

automatic passive protection is provided by means of the four accumulator tanks. The lower plenum must be quickly refilled such that, by the time pumped injection flow reaches the loops, the refilling can be completed and the reflooding of the core can commence. As such, when the reactor is in operation, the accumulators are required to be in their automatic emergency passive mode. Thus, it is only necessary to monitor the tanks sufficiently to assure their operational readiness by verifying that their isolation valves are open and de-energized and their level and pressure are within the safety analysis limits prescribed by the IP-2 Technical Specifications. The current instrumentation is designed to remain functional during normal operating conditions. Installation of additional instrumentation or upgrading the existing instruments, as the technical evaluation report suggests, will not provide any significant safety benefit.

5) Main Steam Flow (3.3.15):

The June 30, 1986 preliminary technical evaluation report incorrectly states that Reg. Guide 1.97 recommends category 1 instrumentation for this variable when, in fact, Table 2 of Reg. Guide 1.97, Rev. 2 actually recommends category 2. We request that the technical evaluation report be corrected accordingly. Also, the requested information for this variable is provided in the revised Table of Variables contained in Attachment 2 to this submittal.

6) Auxiliary Feedwater Flow (3.3.16):

When we originally evaluated this variable, we had used the implementation of NUREG-0737 item II.E.1 as the appropriate guidance to follow and this resulted in classifying it as category 2. As category 2, an evaluation of seismic qualification and redundancy were not applicable. We have re-evaluated this variable and determined that it meets the category 1 requirements of Reg. Guide 1.97, Rev. 2. The attached revised Table of Variables reflects this change.

In the IP-2 design, one motor driven auxiliary feedwater pump (MDAFP) meets the design basis flow requirements of 300 gpm for the first 10 minutes and 400 gpm thereafter. There are two flow instruments per MDAFP, each having a range of 0 to 450 gpm and each connected to a line feeding a steam generator. There are two MDAFPs feeding the four steam generators; thus, since the required heat sink is met with one MDAFP and there are two instruments associated with each MDAFP, the instrumentation is redundant on a per heat sink basis. The flow transmitters were determined to be seismically qualified. Hence, the auxiliary feedwater flow instrumentation is now considered to be Type A, category 1 and any upgrade to install category 1 channels, as requested in the preliminary technical evaluation report is not necessary.

7) Condensate Storage Tank Water Level (3.3.17):

Our original note in the Table of Variables of our August 30, 1985 submittal was intended to convey the commitment that we would seismically qualify one of the two channels since the other was already seismically qualified. It appears that this was not understood in the preliminary technical evaluation report. In any case, we have since evaluated the transmitter and determined that it is also seismically qualified. Therefore, this instrumentation meets category 1 requirements as installed. The revised Table of Variables contained in Attachment 2 reflects this change.

8) Containment Atmosphere Temperature (3.3.19):

This variable is category 3 and has been found acceptable in the technical evaluation report. As such, the current range of 50°F to 150°F is sufficient to cover plant modes that are associated with category 3 normal operating conditions. There are no other design basis conditions that mandate an increased range for this variable as it is not relied on in the EOPs for plant recovery from accidents, as we discussed in our August 30, 1985 submittal. Therefore, modifying this instrumentation to increase its range, as requested in the preliminary technical evaluation report, will not provide any additional safety benefit. As discussed earlier, we are now commiting to upgrade the RHR heat exchanger outlet temperature. Accordingly, note 24 in the Table of Variables has been revised as shown in Attachment 2.

9) Steam Generator Blowdown Radiation (3.3.28):

This variable is not a Type A variable. Since the time of our August 30, 1985 submittal, we have re-evaluated the key variable to detect a Steam Generator Tube Rupture (SGTR) as part of our plant specific implementation and determined that S/G water level narrow range is a more appropriate key variable. The S/G blowdown radiation monitor is isolated on a safety injection signal that would accompany an SGTR, it is therefore of limited use in diagnosing and coping with the accident, and is more appropriately used as a backup variable. When we had selected our Type A variables at the start of our Reg. Guide 1.97, Rev. 2 assesment, we were still developing plant specific EOPs and radiation monitoring betterment programs. In that original assessment we neglected to include the fact that the S/G blowdown radiation monitoring would be isolated on SI rendering it of little use. Therefore, the revised Table of Variables reflects the deletion of S/G blowdown radiation from Type A. The basis of our re-evaluation of SGTR events is provided below.

The SGTR is a small LOCA and the thermal hydraulic behavior of the RCS is similar. The fundamental characteristic of an SGTR is initially an RCS pressure reduction. The water level in the affected S/G may not be significantly greater than that of the intact steam generators prior to safety injection actuation as the normal feedwater control system automatically compensates for changes in steam flow rate and steam generator level due to primary-to-secondary in-leakage via the SGTR.

However, after safety injection actuation, main feedwater is isolated and the feedwater control system can no longer throttle flow to the ruptured S/G, and now, water level in the ruptured S/G does increase significantly due to the mass input via the rupture. Thus, the fundamental characteristic of an SGTR becomes an uncontrolled increase in water level in the ruptured S/G significantly above that of the intact S/Gs. During a SGTR the RCS pressure decreases until low pressurizer pressure causes a reactor trip, safety injection (SI), phase A containment isolation, main feedwater isolation and auxiliary feedwater actuation delivering equal flow to all four S/Gs. Water level in the four S/Gs drops below the narrow range on reactor trip from full power due to the sudden shrinkage of the S/G water mass resulting from the rapid decrease in power. Water level returns to the narrow range significantly earlier in the ruptured S/G since it has mass being introduced into it from the RCS (via the ruptured tube) in addition to auxiliary feedwater whereas the other three intact S/Gs only have auxiliary feedwater flow. During the SGTR the ruptured S/G narrow range level remains significantly higher and increases more rapidly than the other three intact S/Gs. This response of the narrow range water level instrumentation provides confirmation of a steam generator tube rupture event and also identifies the affected steam generator(s). The RCS mass input to the ruptured S/G continues until RCS pressure is below ruptured S/G pressure and SI is terminated. Therefore, S/G water level narrow range instrumentation is the key type A variable and already meets category 1 requirements. Installation of category 1 blowdown radiation monitors is not necessary and will not provide any additional safety benefit. Accordingly, this variable has been deleted from the revised Table of Variables contained in Attachment 2 and our August 30, 1985 commitment to upgrade one channel is no longer necessary.

10) High Level Radioactive Liquid Tank Level (3.3.24) and Radioactive Gas Holdup Tank Pressure (3.3.25):

Although not pointed out in our August 30, 1985 submittal, this instrumentation is accessible on the waste disposal system (WDS) panel under post-accident conditions.

11) Containment Sump Water Temperature (3.3.20):

As discussed in Note 24 of the revised Table of Variables contained in Attachment 2, the containment cooling systems in IP-2 consist of fan cooler units, containment spray, recirculation spray, RHR heat exchangers, SW to the fan cooler units and CCW to the RHR heat exchangers. As indicated earlier and in Note 15, we have re-evaluated the RHR heat exchanger outlet temperature instrumentation and committed to upgrade it to category 2. With that commitment and those discussed in Notes 23 and 24 of the

revised Table of Variables, operation of containment cooling systems is assured in a post-accident condition without reliance on recirculation sump water temperature. Therefore, this variable is not applicable at IP-2 and is so noted in the Table of Variables contained in Attachment 2 to this submittal.

12) Vent From Steam Generator Safety Relief Valves (3.3.29):

The response to this item will be provided by November 30, 1986.

Attachment 2

Revised Table of Variables

September, 1986

## 2.0 TABLE OF VARIABLES

The following terminology and general notes apply for the Indian Point Unit 2 (IP-2) Table of Variables:

### 2.1 TERMINOLOGY

N.R. - Narrow Range
W.R. - Wide Range
HI.R. - High Range
N/R - Not Required
N/A - Not Applicable
N/S - Not Specified
D.F. - Design Flow

### 2.2 GENERAL NOTES

The transmitters' post-accident error (where qualified) is accounted for in the upgraded IP-2 emergency operating procedures (EOP), as applicable. The EOPs will be revised accordingly, depending on final resolution of this evaluation.

See Note 38 in Section 3.0 for details of Class 1E power source.

See Note 39 in Section 3.0 for details of seismic qualification.

The instruments designated as Types A, B and C and Categories 1 and 2 will be specifically identified on the panels so that the operator can easily discern that they are intended for use under accident conditions.

TYPE A

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 / IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| A1 | RCS Pressure, W.R. | 1 | 1 | Yes | Yes | Yes | Plant Specific / 0 to 3000 psig<br>0 to 750 psig | 1E | Yes | Yes | Yes | |
| A2 | RCS Hot Leg Water Temperature | 1 | 1 | Yes | Yes | Yes | Plant Specific / 0 to 700°F | 1E | Yes | Yes | Yes | |
| A3 | RCS Cold Leg Water Temperature | 1 | 1 | Yes | Yes | Yes | Plant Specific / 0 to 600°F | 1E | Yes | Yes | Yes | |
| A4 | Pressurizer Level | 1 | 1 | Yes | Yes | Yes | Plant Specific / 0 to 100% | 1E | Yes | Yes | Yes | |
| A5 | Containment N.R. Pressure | 1 | 1 | Yes | Yes | Yes | Plant Specific / (-5) to +75 psig | 1E | Yes | N/R | N/R | |
| A6 | Steam Generator N.R. Level | 1 | 1 | Yes | Yes | Yes | Plant Specific / 0 to 100% | 1E | Yes | Yes | Yes | |
| A7 | Containment Water Level | 1 | 1 | Yes | Yes | Yes | Plant Specific / See Item B11 | 1E | Yes | Yes | Yes | |
| A8 | Steam Line Pressure | 1 | 1 | Yes | Yes | Yes | Plant Specific / 0 to 1400 psig | 1E | Yes | Yes | Yes | |

TYPE A

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 / IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| A9 | AFW Flow | 1 | 1 | Yes | Yes | Yes | Plant Specific / 0 to 450 gpm | 1E | Yes | Yes | Yes | See September 12, 1986 submittal. |
| A10 | Containment Radiation, HI.R. | 1 | 1 | Yes | Yes | Yes | Plant Specific / 1 to $10^7$ R/hr | 1E | Yes | Yes | Yes | |
| A11 | CST Level | 1 | 1 | Yes | Yes | Yes | Plant Specific / 0 to 45' w.c. | 1E | Yes | Yes | Yes | See September 12, 1986 submittal. |
| A12 | RWST Level | 1 | 2 | Yes | Yes | No | Plant Specific / 0 to 40' w.c. | 1E | Yes | Yes | Yes | Note 3 |
| A13 | Core Exit Temperature | 1 | 3 | No | No | No | Plant Specific / 100°F to 2,200°F | 1E | Yes | Yes | Yes | Note 4 |
| A14 | RCS Subcooling | 1 | 1 | Yes | No | Yes | Plant Specific / -800°F to +800°F | 1E | Yes | Yes | Yes | Note 5 and September 12, 1986 submittal |

TYPE B

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| B1 | Neutron Flux | 1 | 3 | N/A | N/A | N/A | $10^{-6}$ to 100% Full Power | $10^{-9}$ to 120% Full Power | 1E | Yes | Yes | Yes | Note 7 and September 12, 1986 submittal |
| B2 | Control Rod Position | 3 | 3 | N/R | N/R | N/R | Full In or Not Full In | 0 to 144" | 1E | Yes | N/R | N/R | |
| B3 | RCS Soluble Boron Concentration | 3 | 3 | N/R | N/R | N/R | 0 to 6,000 ppm | 60 to 6,000 ppm | 1E | No | N/R | N/R | Note 8 |
| B4 | RCS Cold Leg Water Temperature | 3 | 1 | Yes | Yes | Yes | 50°F to 400°F | 0 to 600°F | 1E | Yes | Yes | Yes | |
| B5 | RCS Hot Leg Water Temperature | 1 | 1 | Yes | Yes | Yes | 50°F to 750°F | 0 to 700°F | 1E | Yes | Yes | Yes | Range Satisfies Reg. Guide 1.97 Rev. 3 |
| B6 | RCS Cold Leg Water Temperature | 1 | 1 | Yes | Yes | Yes | 50°F to 750°F | 0 to 600°F | 1E | Yes | Yes | Yes | Note 9 |
| B7 | RCS Pressure | 1 | 1 | Yes | Yes | Yes | 0 to 3,000 psig | 0 to 3,000 psig<br>0 to 750 psig | 1E | Yes | Yes | Yes | Note 10 |

TYPE B

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| B8 | Core Exit Temperature | | 3 | 3 | N/R | N/R | N/R | 200°F to 1,650°F | 100°F to 2,200°F | 1E | Yes | Yes | Yes | |
| B9 | Coolant Level in Reactor | | 1 | 1 | No | Yes | Yes | Bottom of Core to Top of vessel | 0-120% | 1E | Yes | Yes | Yes | Note 11 |
| B10 | RCS Degrees of Subcooling | | 2 | | | | | 200°F Subcooling to 35°F Superheat | | | | | | Refer to item A14 |
| B11 | Containment Sump Water Level | N.R. | 2 | 1 | Yes | Yes | Yes | Narrow Range: Sump Wide Range: Bottom of containment to 600,000 gal. level equivalent | | 1E | Yes | Yes | Yes | Range satisfies Reg. Guide 1.97 Rev. 3 |
| | | W.R. | 1 | 1 | Yes | Yes | Yes | Bottom of Sump to Design Basis Flood level | | | | | | |
| B12 | Containment N.R. Pressure | | 1 | | | | | 0 to Design Pressure | | | | | | Refer to Item A5 |
| B13 | Containment Isolation Valve Position | | 1 | 2 | Yes | Yes | N/A | Closed - Not Closed | Closed - Not Closed | 1E | No | N/R | N/R | Note 12 |

TYPE C

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 / IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| C1 | Core Exit Temp. | 1 | | | | | 200°F to 1,650°F | | | | | Refer to Item A13 |
| C2 | Radioactivity in RCS | 1 | 3 | N/A | N/A | N/R | 1/2 Tech. Spec. Limit to 100 Times T.S. Limit<br>1 uci/g to 10 ci/g | 1E | No | N/R | N/R | Note 13 |
| C3 | Analysis of Primary Coolant | 3 | 3 | N/R | N/R | N/R | 10 uCi/g to 10 Ci/g<br>10 uCi/g to 10 Ci/g | 1E | No | N/R | N/R | Note 14 |
| C4 | RCS W.R. Pressure | 1 | | | | | 0 to 3,000 psig | | | | | Refer to Item B7 |
| C5 | Containment N.R. Pressure | 1 | | | | | 10 psia to design pressure | | | | | Refer to Item A5 |
| C6 | Containment Sump Water Level N.R.<br>W.R. | 2<br>1 | | | | | Narrow Range: Top to Bottom<br>Wide Range: Plant Specific | | | | | Refer to Item B11 |

TYPE C

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 / IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| C7 | Containment Area Radiation | 3 | | | | | 1 R/hr to $10^4$ R/hr | | | | | Refer to Item A10 |
| C8 | Effluent Radioactivity from condenser Air Ejector | 3 | | | | | $10^{-6}$ to $10^{-2}$ $\frac{uCi}{cc}$ | | | | | Refer to Item E6 |
| C9 | Containment Hydrogen Concentration | 1 | 1 | Yes | Yes | Yes | 0 to 10% / 0 to 10% | 1E | Yes | Yes | Yes | |
| C10 | Containment Pressure | 1 | 1 | Yes | Yes | Yes | 10 psia to 3x Design Pressure / -10 psig to 150psig | 1E | Yes | Yes | Yes | |
| C11 | Containment Effluent Radio-Activity | 2 | | | | | $10^{-6}$ to $10^{-2}$ $\frac{uCi}{cc}$ | | | | | Refer to Item E7 |
| C12 | Effluent Radio-Activity From Buildings | 2 | | | | | $10^{-6}$ to $10^{3}$ $\frac{uCi}{cc}$ | | | | | Refer to Item E7 |

TYPE D

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| D1 | RHR System Flow | 2 | 2 | Yes | N/R | N/R | 0 to 110% D.F. | 0 to 7000 gpm | 1E | Yes | Yes | Yes | |
| D2 | RHR Heat Exchanger Outlet Temperature | 2 | 3 | No | N/R | N/R | 32° to 350°F | 75° to 400°F | 1E | Yes | Yes | Yes | Note 15 and September 12, 1986 submittal. |
| D3 | Accumulator Tank Level | 2 | N/A | | | | 10 to 90% volume | | | | | | September 12, 1986 submittal |
| D4 | Accumulator Tank Pressure | 2 | N/A | | | | 0 to 750 psig | | | | | | September 12, 1986 submittal |
| D5 | Accumulator Isolation Valve Position | 2 | N/A | | | | Closed or Open | | | | | | September 12, 1986 submittal |
| D6 | Boric Acid Charging Flow | 2 | N/A | | | | 0 to 110% Design Flow | | | | | | Note 17 |
| D7 | Flow in HPI System | 2 | 2 | Yes | N/R | N/R | 0 to 110% Design Flow | 0 to 600 gpm | 1E | Yes | N/R | N/R | |
| D8 | Flow in LPI System | 2 | | | | | 0 to 110% Design Flow | | | | | | Refer to D1 |

TYPE D

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| D9 | RC Pump Status | 3 | 3 | N/R | N/R | N/R | Motor Current | 0 to 800 amp | 1E | Yes | N/R | N/R | |
| D10 | Pressurizer Safety and Relief Valve Position | 2 | 2 | Yes | N/R | N/R | Closed - Not Closed | Closed - Not Closed | 1E | Yes | N/R | N/R | |
| D11 | Pressurizer Heater Status | 2 | 3 | N/A | N/R | N/R | Current | Breaker Position | 1E | Yes | N/R | N/R | Note 18 |
| D12 | Quench Tank (Pressurizer Relief Tank) Level | 3 | 3 | N/R | N/R | N/R | Top to Bottom | 0 to 100 in Water | 1E | Yes | Yes | Yes | Note 19 |
| D13 | Quench Tank (PRT) Pressure | 3 | 3 | N/R | N/R | N/R | 0 to Design Press. | 0 to 100 psig | 1E | Yes | Yes | Yes | |
| D14 | Quench Tank (PRT) Temperature | 3 | 3 | N/R | N/R | N/R | 50° to 750°F | 50° to 300°F | 1E | Yes | N/R | N/R | Note 20 |
| D16 | Steam Generator Safety/Relief Valve Position or Main Steam Flow | 2 | 2 | Yes | N/R | N/R | Closed - Not Closed | 0 to $4x10^6$ lbs/hr. | 1E | Yes | N/R | N/R | Note 21 |
| D17 | Main FW Flow | 3 | 3 | N/R | N/R | N/R | 0 to 110% D.F. | 0 to 4 x $10^6$ $\frac{LB}{hr}$ | 1E | Yes | Yes | Yes | |

TYPE D

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| D18 | Containment Spray Flow | 2 | 2 | Yes | N/R | N/R | 0 to 110% D.F. | 0 to 2400 gpm | 1E | Yes | N/R | N/R | Note 22 |
| D19 | Heat Removal By The Fan-Cooler Units | 2 | 3 | No | N/R | N/R | Plant Specific | See Note 23 | 1E | Yes | N/R | N/R | Note 23 |
| D20 | Containment Atmosphere Temperature | 2 | 3 | No | N/R | N/R | 40°F to 400°F | 50°F to 150°F | 1E | Yes | Yes | Yes | Note 24 and September 12, 1986 submittal |
| D21 | Containment Sump Water Temperature | 2 | N/A | | | | 50°F to 250°F | | | | | | Note 25 and September 12, 1986 submittal |
| D22 | CVCS Make-up Flow-In | 2 | 3 | No | N/R | N/R | 0 to 110% D.F. | 0 to 120 gpm | 1E | Yes | N/R | N/R | Note 26 |
| D23 | CVCS Letdown Flow-Out | 2 | 3 | No | N/R | N/R | 0 to 110% D.F. | 0 to 125 gpm | 1E | Yes | N/R | N/R | Note 26 |
| D24 | Volume Control Tank Level | 2 | 3 | No | N/R | N/R | Top to Bottom | 0 to 100% | 1E | Yes | N/R | N/R | Note 26 |
| D25 | Component Cooling Water (CCW) Temperature to ESF System | 2 | 3 | No | N/R | N/R | 32° to 200°F | 50° to 200°F | 1E | Yes | N/R | N/R | Note 27 |

TYPE D

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| D26 | CCW Flow to ESF Systems | 2 | 3 | No | N/R | N/R | 0 to 110% D.F. | 2000 to 12,000 gpm | 1E | Yes | N/R | N/R | Note 27 |
| D27 | High Level Radioactive Liquid Tank Level | 3 | 3 | N/R | N/R | N/R | Top to Bottom | 0 to 150 in Water | Pneumatic | No | N/R | N/R | Note 28 and September 12, 1986 submittal |
| D28 | Radioactive Gas Hold-up Tank Pressure (Gas Decay Tanks) | 3 | 3 | N/R | N/R | N/R | 0 to 150% D.P. | 0 to 150 psig | Pneumatic | No | N/R | N/R | Note 29 and September 12, 1986 submittal |
| D29 | Emergency Ventilation Damper Position | 2 | 2 | Yes | N/R | N/R | Open - Close | Open - Close | 1E | Yes | N/R | N/R | |
| D30 | RWST Level | 2 | 2 | Yes | Yes | N/R | Top to Bottom | 0 to 40" W.C. | 1E | Yes | Yes | Yes | |
| D31 | Pressurizer Level | 1 | 1 | Yes | Yes | Yes | Bottom to Top | Tap to Tap | 1E | Yes | Yes | Yes | Note 30 |
| D32 | Steam Generator Level | 1 | | | | | From tube sheet to separators | | | | | | Refer to A6 |

TYPE D

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| D33 | S.G. Pressure (Steam Line Pressure) | 2 | | | | | From Atmospheric Pressure to 20% Above the Lowest Safety Valve Setting | | | | | | Refer to A8 |
| D34 | APW Flow | 2 | 2 | Yes | N/R | N/R | 0 to 110% Design Flow | 0 to 450 gpm | 1E | Yes | Yes | Yes | |
| D35 | CST Level | 1 | | | | | Plant Specific | | | | | | Refer to A11 |
| D36 -1 | Status of Standby Power (480 Vac, Buses, Inverter Output Buses) | 2 | 2 | Yes | Yes | N/R | Voltages, Currents | Voltage Current, Megawatt | 1E | Yes | N/R | N/R | |
| D36 -2 | Status of Standby Power (6.9 kV Buses) | 2 | 3 | N/A | N/R | N/R | Voltages, Currents | Voltage Current, Megawatt | 1E | No | N/R | N/R | See Note 31 |
| D36 -3 | Status of Instrument Air System | 2 | 2 | Yes | N/R | N/R | Pressure | 0-160 psig | Pneumatic | Yes | N/R | N/R | No Hydraulic Energy Sources are in IP-2 Design |

## TYPE E VARIABLES

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| E1 | Radiation Exposure Rate (Inside Bldgs. or Areas, Where Access is Req'd to Service Equipment Important to Safety) | 3 | 3 | N/R | N/R | N/R | $10^{-2}$ to $10^{4}$ R/hr | $10^{-4}$ to 10 R/hr | 1E | Yes | Yes | Yes | Note 32 |
| E2 | Containment Area Radiation | 1 | | | | | 1 to $10^{7}$ R/hr | | | | | | Refer to Item A10 |
| E3 | Containment or Purge Effluent (Airborne Radioactivity Noble Gas) | 2 | | | | | $10^{-6}$ to $10^{5}$ $\frac{uCi}{cc}$ | | | | | | Refer to Item E7 |
| E4 | Reactor Shield Building Annulus-Noble Gas | | | | | | $10^{-6}$ to $10^{4}$ $\frac{uCi}{cc}$ | | | | | | Not in IP2's Design |
| E5 | Noble Gases Released From PAB, Including Waste Gas Decay Tank | 2 | | | | | $10^{-6}$ to $10^{3}$ $\frac{uCi}{cc}$ | | | | | | Refer to Item E7 |
| E6 | Condenser Air Ejector Radiation-Noble Gas | 2 | 3 | No | N/R | N/R | $10^{-6}$ to $10^{5}$ $\frac{uCi}{cc}$ | $10^{-6}$ to $10^{-3}$ $\frac{uCi}{cc}$ | 1E | Yes | Yes | Yes | Note 33 |

## TYPE E VARIABLES

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOP | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| E6-1 | Condenser Air Ejector Flow | | 2 | N/A | N/A | N/A | N/A | 0-110% Design Flow | | N/A | N/A | N/A | N/A | The design flow of 20 CFM will be applied to release assessment. |
| E7 | Noble Gases Released | RAD | 2 | 2 | Yes | N/R | N/R | $10^{-6}$ to $10^{4}$ $\frac{uCi}{cc}$ | $10^{-7}$ to $10^{5}$ $\frac{uCi}{cc}$ | 1E | Yes | Yes | Yes | Note 34 |
| | Thru Common Plant Vent | FLOW | 2 | 3 | No | N/A | N/R | 0 to 110% | 0 to 170,000 cfm | 1E | Yes | Yes | Yes | Note 34 |
| E8 | Vent From SG Safety and Atmosphere Dump Valves-Noble Gas | | 2 | N/A | | | | $10^{-1}$ to $10^{3}$ $\frac{uCi}{cc}$ | | | | | | Note 35 and September 12, 1986 submittal |
| E9 | All Other Identified Release Points -Noble Gas | | 2 | | | | | $10^{-6}$ to $10^{2}$ $\frac{uCi}{cc}$ | | | | | | Note 36 |
| E10 | Airborne Rad. Part & Halogens From All Identified Points | | 3 | 3 | N/R | N/R | N/R | $10^{-3}$ to $10^{2}$ $\frac{uCi}{cc}$ | $10^{-3}$ to $10^{2}$ $\frac{uCi}{cc}$ | 1E | N/A | N/A | N/A | See E7 for Flow Requirements and Compliance |

## TYPE E VARIABLES

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| E11 | Airborne Radio-Halogens and Particulates (Portable Sampling with On-site Analysis Capability) | 3 | 3 | N/R | N/R | N/R | $10^{-9}$ to $10^{3}$ uCi/cc | $10^{-9}$ to $10^{3}$ uCi/cc | N/A | N/A | N/A | N/A | |
| E12 | Plant and Environs Radiation. (Portable Instrumentation) | 3 | 3 | N/R | N/R | N/R | $10^{-3}$ to $10^{4}$ R/hr<br>$10^{-3}$ to $10^{4}$ rads/hr | Various H.P. Instruments Meet Ranges | N/A | N/A | N/A | N/A | |
| E13 | Plant and Environs Radioactivity (Portable Instrumentation) | 3 | 3 | N/R | N/R | N/R | Multi-Channel Gamma-Ray Spectrometer | Multi-Channel Gamma-Ray Spectrometer | N/A | N/A | N/A | N/A | |
| E14 | Wind Direction | 3 | 3 | N/R | N/R | N/R | 0 to 360° | 0 to 540° | 1E | Yes | Yes | Yes | |
| E15 | Wind Speed | 3 | 3 | N/R | N/R | N/R | 0 to 30 MPS | 0 to 56 MPS | 1E | Yes | Yes | Yes | |
| E16 | Estimation of Atm. Stability (Vertical Temperature Difference) | 3 | 3 | N/R | N/R | N/R | -9°F to (+18°)F | -10°F to (+10°)F | 1E | Yes | Yes | Yes | Note 37 |

## TYPE E VARIABLES

IP2 Summary Information Sheet For RG 1.97 Rev. 2

| Item | Variable | CATEGORY R.G. 1.97 | CATEGORY IP-2 | Environmental Qualification | Seismic Qualification | Redundancy | RANGE R.G. 1.97 | RANGE IP-2 | Power Source | CCR Display | TSC | EOF | Comments |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| E17 | Accident Sampling Capability - Analysis On-site | 3 | 3 | N/R | N/R | N/R | Grab Sample | Grab Sample | 1E | N/A | N/A | N/A | |

## 3.0 NOTES AND JUSTIFICATION

Note 1: Deleted

Note 2: Deleted

Note 3: Upgrade is required to meet Category 1 requirements. A second qualified channel will be installed.

Note 4: The system will be upgraded to meet the intent of R.G. 1.97 and NUREG-0737 for instrumentation to monitor core cooling. (See Appendix I for details).

Note 5: The inputs to the existing subcooling margin monitor and SAS/SPDS are redundant and qualified, the monitors and the computers are not seismically qualified. During and after a seismic event the qualified wide range RCS pressure and temperature instrumentation can be used. The subcooling margin monitor, added in good faith to meet post-TMI NUREG-0578 requirements, is considered adequate. (See Appendix I for details). The SAS/SPDS will be operational by April, 1988 in accordance with our confirmatory order revision.

Note 6: Deleted. See September 12, 1986 submittal.

Note 7: At IP-2 this is a category 3 variable. The conclusion that the existing equipment is adequate is based on the fact that upon a reactor trip, the nuclear instrumentation will provide an accurate indication of the shutdown status of the core before a "harsh" post-accident containment environment develops. In the longer term, to assure no return to power, boron sampling analysis is adequate for confirming proper boron concentration to ensure that the reactor is shutdown. Con Edison has made substantial plant modifications to implement the upgraded Post Accident Sampling System (PASS) in response to NUREG-0737 items II.B.3 and II.F.1. The key variable to determine the state of the core in the long term recovery is Core Exit Temperature (CET), which we are now proposing to upgrade. Thus, during the period of time required to obtain and analyze a coolant sample our proposed CET instrumentation will be available, and through the upgraded Emergency Operating Procedures (EOPs), the operator will be directed to take action to assure adequate core cooling. The actions to adequately cool the core use borated sources of water, as appropriate, thereby ensuring adequate shutdown status at the same time.

Note 8: Two displays are available; one locally in the High Radiation Sampling System (HRSS) room and another in the radiochem/counting room. This is Category 3 instrumentation and 6.2(g) of NUREG-0737, Supplement 1 accepts displays in locations other than the CCR.

Note 9: Existing instrument range will be upgraded to 0-700°F to meet the Regulatory Guide 1.97, Rev. 3 specified range

Note 10: One of the two existing recorders does not meet the range. An upgrade will be implemented to meet the required range.

Note 11: The original transmitters are being replaced with environmentally qualified transmitters during the 1986 refueling outage. All other components of the system are qualified.

Note 12: This variable serves as a backup variable in monitoring containment integrity. Containment pressure is the primary variable. Therefore, it is considered that Category 2 is more appropriate for valve position. Grouped position indication in the CCR for the containment isolation valves in the containment sampling system will be provided.

Note 13: This variable serves as a backup variable for monitoring fuel cladding breach. The methods include the delayed neutron gamma monitor and grab sampling for analysis. The key variable that it backs up is core exit temperature, which will be upgraded. Therefore, it is considered appropriate that this variable be classified as Category 3. The existing instrumentation meets the requirements of Category 3. The readout is available in the radiochem/counting room.

Note 14: A Canberra isotopic analyzer is located in the waste gas compressor room. Its readout is available in the radiochem/counting room. This is Category 3 instrumentation and 6.2(g) of NUREG-0737, Supplement 1 accepts displays in locations other than the CCR.

Note 15: RHR heat exchanger outlet temperature will be upgraded to meet category 2 requirements.

Note 16: Deleted

Note 17: This variable is not applicable to IP2's design. At IP2, the Operator has sufficient information to indicate boric acid flow into the RCS, viz. RWST level, discharge valve position, and status of the Safety Injection System. This instrumentation meets Category 2 requirements.

By letter dated August 2, 1985, Con Edison proposed removal of the entire Boron Injection Tank package, since sufficient margin exists within the RWST.

Note 18: Existing instrumentation monitors breaker position. An upgrade to install heater current monitoring instrumentation, qualified to Category 2 requirements, will be implemented.

Note 19: The existing range of 0-100 in. covers all anticipated operational occurrences.

Note 20: The range of existing instrumentation will be upgraded to 50-350$^{o}$F, which covers saturation temperature of 338$^{o}$F at tank design pressure of 100 psig.

Note 21: Indian Point 2 relies on Main Steam Line Flow (Full Range), Steam Generator Pressure, and physical verification (audible and visible) of valve actuation. The information provided in the table is for Main Steam Line Flow.

Note 22: The existing flow instrumentation range of 0-2400 gpm covers the maximum possible flow during the recirculation phase and satisfies all anticipated post-accident conditions.

Note 23: IP2 monitors fan cooler unit heat removal from the containment by observation of a) the cooling water outlet flow, b) the fan motor status, and c) the fan-cooler dampers' position. The cooling water outlet flow transmitters will be upgraded to meet qualification requirements of Category 2.

Note 24: This variable is not essential for IP-2 post accident recovery and is not relied on in our EOPs. The purpose of this variable is to verify adequate containment heat removal. In IP-2 heat removal is accomplished by the Fan Cooler Units (FCUs) and Containment Spray (CS). By observing Service Water (SW) flow, Component Cooling Water (CCW) flow and RHR heat exchanger outlet temperature, adequate containment heat removal by the FCUs and CS is verified. In addition, containment pressure trending downward will also confirm heat removal. We are now proposing to upgrade the SW instrumentation as discussed in note 23, CCW flow and inlet temperature to the RHR heat exchanger and RHR heat exchanger outlet temperature. Thus the improved instrumentation will assure valid indication of containment heat removal.

Note 25: This variable is not applicable at IP-2. The NPSH calculations for IP-2 conservatively assume saturated conditions.

Note 26: The CVCS system (charging, letdown, volume control tank, etc.) is not a safety grade system and is not required for recovery at IP-2. The charging pumps are removed from the 1E buses after a SI signal and concurrent loss of off-site power. The system will be used, if available, but if not, the safety systems are relied on for recovery. Therefore, the CVCS flow and tank level instrumentation is considered to be Category 3. Also, if necessary, the charging pump running status lights in the CCR can be used to indicate their condition.

Note 27: The system will be upgraded with a qualified sensor to meet the Category 2 requirements. Present range meets all operational requirements.

Note 28: The range of level measurement of this tank (the Waste Hold-Up Tank) covers all anticipated operational occurrences. The display is located on the Waste Disposal System (WDS) panel. The instrumentation is Category 3 and Section 6.2 (g) of NUREG-0737 Supplement 1 accepts displays in locations other than the CCR.

Note 29: The existing instrumentation covers 0-150 psig. This range covers all anticipated operational occurrences, since the waste gas compressor design pressure is 150 psig and the tank design pressure is 150 psig. Additionally, safety valves are set at 110 psig. The display location in the WDS panel is considered adequate. (See Note 28).

Note 30: Existing instrumentation range of 0 to 100% span covers 85% of total volume (tap-to-tap). Considering this large fraction and the severe non-linearity outside the tangent points, we have concluded that the present installation meets the intent of the R.G.

Note 31: There is no display in the Central Control Room (CCR) to monitor the alternate off-site source load on the 13.8/6.9 KV-20 MVA autotransformer. An ammeter will be provided to indicate the status of this alternate standby power supply. This instrument will be used by the Operator for diagnostic purposes of the alternate offsite source loading which is non-1E. Therefore, this instrument is considered to be Category 3.

Note 32: An area radiation monitor will be installed in the necessary breaker service access area. The range of the existing IP2 instruments is based on the fact that it covers all anticipated occurrences. In addition, the areas where exposure rate might exceed one R/hr are locked to preclude entry. If entry is needed, a health physics technician will monitor the area prior to access.

Note 33: The existing channel meets the requirements for Type C Category 3. However, the monitor will be replaced to meet the criteria for Category 2 and will meet the range requirements for Type E.

Note 34: Containment purge and PAB ventilation system discharge is through the common plant vent. The flow instrumentation satisfies the intent of the R.G. The system is designed so that in the event of its failure, the operator will set the microprocessor to a predetermined, conservative value of flow release rate which is at least as high as the value expected during all modes of system operation.

Note 35: The IP2 method for estimating releases through the atmospheric dump valves and the steam generator safety valves during the course of an accident is to obtain samples, taken upstream of the main steam isolation valves, of the entrapped noble gases for analysis in the on-site radiochem lab. Combining this information with total steam flow from the existing low range flow meters will yield the required data on quantity of radioactivity releases. Additionally, the FSAR, Sect. 14.2, conservatively concludes that a SG tube rupture would not result in significant radiation exposure to plant workers and members of the public.

Note 36: There are no other unmonitored release points.

Note 37: The existing system for estimation of atmospheric stability based on vertical temperature difference covers all anticipated conditions and all the historic atmospheric variations of the meteorological conditions in the plant vicinity.

Note 38: Power Supply (1E)

Qualification: The level of environmental qualification of IP2 power source safety related equipment (such as station instrument 118VAC vital buses and 480VAC MCC with safeguards and associated cable loads) is described in FSAR Sections 7.1.4 and 7.1.3.3.3.

Separation Criteria: The IP-2 electric systems do not conform to the recommendations contained in Reg. Guide 1.75 since this guide was not part of the original criteria for the design and construction of IP2. Proposed modifications will be integrated within our existing separation provisions.

Note 39: Active instrument channel components, including transmitter or sensor, and rack mounted signal conditioning equipment are seismically qualified. Control board indicators will be upgraded for seismic qualification. Containment isolation valve or damper limit switches and related active components are seismically qualified. The cable raceway system has been analyzed for seismic adequacy for the IP-2 DBE in the Indian Point Probabilistic Safety Study. The seismic design of the control boards is discussed in Section 7.2.1.11 of the IP2 FSAR, Rev. 2.