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SUBJECT: Forwards response to License Condition 28 D addressing currently available analyses for inadequate core cooling.No addl instrumentation necessary to meet license condition or TMI Item II.F.2 requirement.

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AUG 31 1982

Mr. A. Schwencer, Chief
Licensing Branch No. 2
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO LICENSE CONDITION #28D
ER 100450 FILE 841-2
PLA-1267

Docket No. 50-387

Dear Mr. Schwencer:

The attached report addresses currently available BWR Owners' Group (BWROG) analyses for inadequate core cooling. This report is required by part i of license condition #28D. The BWROG analyses will be documented in two reports. One (on reactor water level instrumentation) has already been transmitted to NRC by a letter from T. J. Dente to H. R. Denton on August 13, 1982. The other (on inadequate core cooling (ICC)) is expected to be transmitted by October, 1982.

The attached report addresses the BWROG analyses and is based on the final water level report and the most recent draft of the ICC report. The attached report concludes that the existing Susquehanna design is adequately reliable to provide unambiguous information on core cooling to the reactor operator. We believe that no additional instrumentation is necessary to meet this license condition or requirement II.F.2 in NUREG-0737. Therefore this submittal completes our action on this issue. We request your concurrence on this matter.

Very truly yours,

N W Curtis

N. W. Curtis
Vice President-Engineering & Construction-Nuclear

DPM/mks

Attachment

cc: R. L. Perch - NRC

B001

RESPONSE TO LICENSE CONDITION 28D

Introduction

In NUREG-0737, NRC established a requirement for licensees to identify additional instrumentation to provide unambiguous indication of inadequate core cooling (ICC). Regulatory Guide 1.97, Revision 2 indicated that in-core thermocouples would satisfy this requirement. Utilities attempted to persuade the NRC that additional instrumentation was unnecessary and challenged the usefulness of in-core thermocouples. In an attempt to resolve the issue, utilities and NRC agreed to a BWR Owners' Group (BWROG) study to specifically address ICC concerns. The study was to evaluate means of providing reliable information to detect: the approach towards ICC, the existence of ICC, and the return to adequate core cooling. The study was to consider local and core-wide ICC, the reliability of existing instrumentation, and the impact of additional instrumentation.

The BWROG first evaluated the relationship between reactor water level and adequate core cooling. In order to clearly demonstrate that reactor water level is a viable indicator of ICC and due to the complexity of the issue, the BWROG scoped work into two activities. The first was to evaluate the reliability of existing BWR reactor water level measurement systems. The second was a study of ICC. The report resulting from the first part was transmitted to NRC in a letter from T. J. Dente to H. R. Denton on August 13, 1982. The report resulting from the second part is in draft form and is anticipated to be transmitted to NRC by October, 1982. These reports provide substantial evidence to conclude that reactor water level is the most suitable parameter for operational control to avoid and mitigate ICC.

Reactor Water Level Instrumentation Report

The BWROG studied four types of reactor water level instrumentation which are representative of existing designs. The conclusion of the study was that the instrumentation used

> "through many years of operating experience have demonstrated very high degrees of capability to provide required information in various conditions of reactor operation. Almost without exception, the information presented to the operator is not ambiguous, and trips, initiations, and other signals taken from the level measurement systems have occurred as required."

The report goes on however to note a few reported events resulting in spurious signals and erroneous information to the operator, none of which resulted in serious consequences. The report indicates the desirability of an overall reassessment of the level system vulnerabilities against a list of potential areas of improvement. The report concludes that

> "no modifications should be made to any specific system until a thorough plant specific analysis is conducted. Interaction of systems in a specific plant design can significantly affect the degree of design change necessary to improve a system and may possibly demonstrate that a design change is not required."

Inadequate Core Cooling Report

The following concepts are extracted from Revision 2 of the draft report on ICC, as amended by committee meeting on August 2 and 3, 1982.

1) Definition of ICC in terms of fuel and clad peak temperatures.
Clad temperatures in the range of 1300°F to 1500°F may likely result in the release of gaseous fission products in the fuel to clad gap by means of perforation produced by weakening of the fuel cladding. At temperatures in excess of 1800°F, clad metal-water chemical heat reaction commences and accelerates the heat rate. The report suggests that ICC might be defined as reaching peak temperatures between 1300°F and 1800°F in an average fuel bundle.

2) Operating states which might lead to ICC.
The relationship among reactor power, coolant inventory (water level), and recirculation flow which results in ICC is developed. The most extensive development of ICC results from operation at critical heat flux within the normal operating power range. Critical heat flux is treated extensively in present safety analyses and its occurrence is prevented by a substantial regulatory methodology including power-flow trip lines, limiting power distribution, and reactor trip systems. This rationale is extended down to zero flow and zero power including ICC conditions which may accompany high void fraction pumped recirculation flow.

From the above, the report concludes that the ICC requirements of NUREG 0737, item II.F.2 and Regulatory Guide 1.97 were not meant to be applicable to normal power range operation critical heat flux conditions, but apply to BWR's only at decay power conditions.

3) Water level as an indicator of ICC.
Applying only decay power conditions, a scenario was developed based on reactor scram, recirculation pump trip, reactor pressure vessel isolation, and loss of all makeup water systems (safety and non-safety). The steam produced by sensible and decay heat is assumed to be lost from the reactor pressure vessel at constant pressure. The time history of water level in this condition is shown in Figure 1. The relationship between water level and peak cladding temperature (which is an indicator of ICC) is shown in Figure 2. Sensitivity of this relationship to core uncovery times is shown to be very flat (see Figure 3). The assumption of a constant pressure (1,000 psia) was shown to be conservative as compared to a similar scenario at low pressure (100 psia) and a saw tooth shaped pressure function indicative of periodic safety/relief valve operation.

Accordingly it is concluded that reactor vessel water level is a valid indicator of ICC including approach to, existence of, and return from those conditions.

4) Local versus global detection of ICC.
A literature review indicated that core damage will not propagate once the core is recovered with water. A scenario is postulated that results in local fuel damage during the existence of global ICC, where the blockage prevents subsequent cooling of the damaged channel. Damage propagation subsequent to global ICC recovery will be restricted to those bundles where sufficient fuel damage occurred during the global ICC to totally cut off the bundle water flow after recovery.

The use of instrumentation to detect this existence of local ICC was considered and rejected because bundle damage sufficient to cause complete blockage of cooling subsequent to recovery would also destroy any instrument placed therein.

5) Additional Instrumentation.
In addition to water level, there are a number of other existing instrument systems which provide information relative to the question of ICC. These include core spray flow rate, flows to and from the reactor vessel, primary containment radiation levels and hydrogen concentration levels, and activity sampling in reactor coolant water and the suppression pool.

6) Risk significance of ICC.
The contribution of water level measurement system failure to core melt probability was evaluated based on modifying an existing PRA for a BWR-4 plant with MARK II containment. The basic approach was to modify the event trees to identify the risk contributed by the water level system. Major concerns considered were: loss of level indication due to loss of reference leg under high drywell temperature and low vessel pressure conditions; concurrent or common failures of level instruments, and reference leg breaks. The results are considered to be representative of the Susquehanna design.

It is shown that water level measurement failures contribute less than 13% of the overall probability of core melt. Improvements in the level measurement system can reduce the contribution of level instrument failure to overall risk down to 3%. These improvements include reduction or mitigation of errors caused by high drywell temperatures, validation of level signals, and increasing the probability of timely ADS operation by manual actuation. Susquehanna has combined elements of these improvements in its design including reduced and equal vertical drops within primary containment for both the reference and variable legs of the multiple instrument channels which mitigate the effects of high drywell temperatures. In addition, the Susquehanna Emergency Operating Procedures (which are based on the BWROG Emergency Procedure Guidelines) provide assurance of timely manual ADS operation.

7) Cost/Benefit of Additional Instrumentation.
An evaluation of alternative or diverse means of detecting ICC was conducted. Thirty three concepts, listed in Table 1 were evaluated with many of these concepts being discarded after the preliminary evaluation. Finally, four devices were selected for further evaluation of performance and cost. These devices included: in-core thermocouples in the LPRM tubes; heated junction thermocouples as a point level measurement inside the LPRM tubes; steam dome thermocouples; source range monitors as an ICC detection device. A cost/benefit analyses was performed on these instrument system additions using a technique proposed in SECY-81-513 "Plan for Early Recognition of Safety Issues: August 25, 1981. The technique uses the following formula to evaluate a priority score.

$$S = \frac{\text{Safety Benefit}}{\text{Cost}} = \frac{N\Delta(FR)R^{0.2}}{C + NI}$$

Where:

S = Priority score

N = Number of reactors affected

R = Consequence, in curies released

$R^{0.2}$ = Weighting Factor

F = Event frequency in events per reactors year

C = Forward looking NRC cost in millions of dollars

I = Forward looking industry cost in millions of dollars per reactor

Δ = Mathematical operator to indicate the change in the quantity within the brackets.

Priorities are assigned to the score in rough boundaries as follows:

S < 1,000.....Low Priority

1,000 < S < 10,000.....Medium Priority

10,000 < S.....High Priority

These ranges were developed based on an evaluation of light water reactor safety issue priority lists and are included in EPRI report EPRI.RP-1585 June 3, 1982. The alternative ICC detection devices were evaluated as above with the assumptions that they increased the probability of recognition of impending ICC by a factor of 5, and that the utility cost would be 3 million dollars. The priority score was calculated to be 35 with a possible range of 3 to 400. This priority score can be considered to fall in the Low Priority range compared to other LWR safety issues.

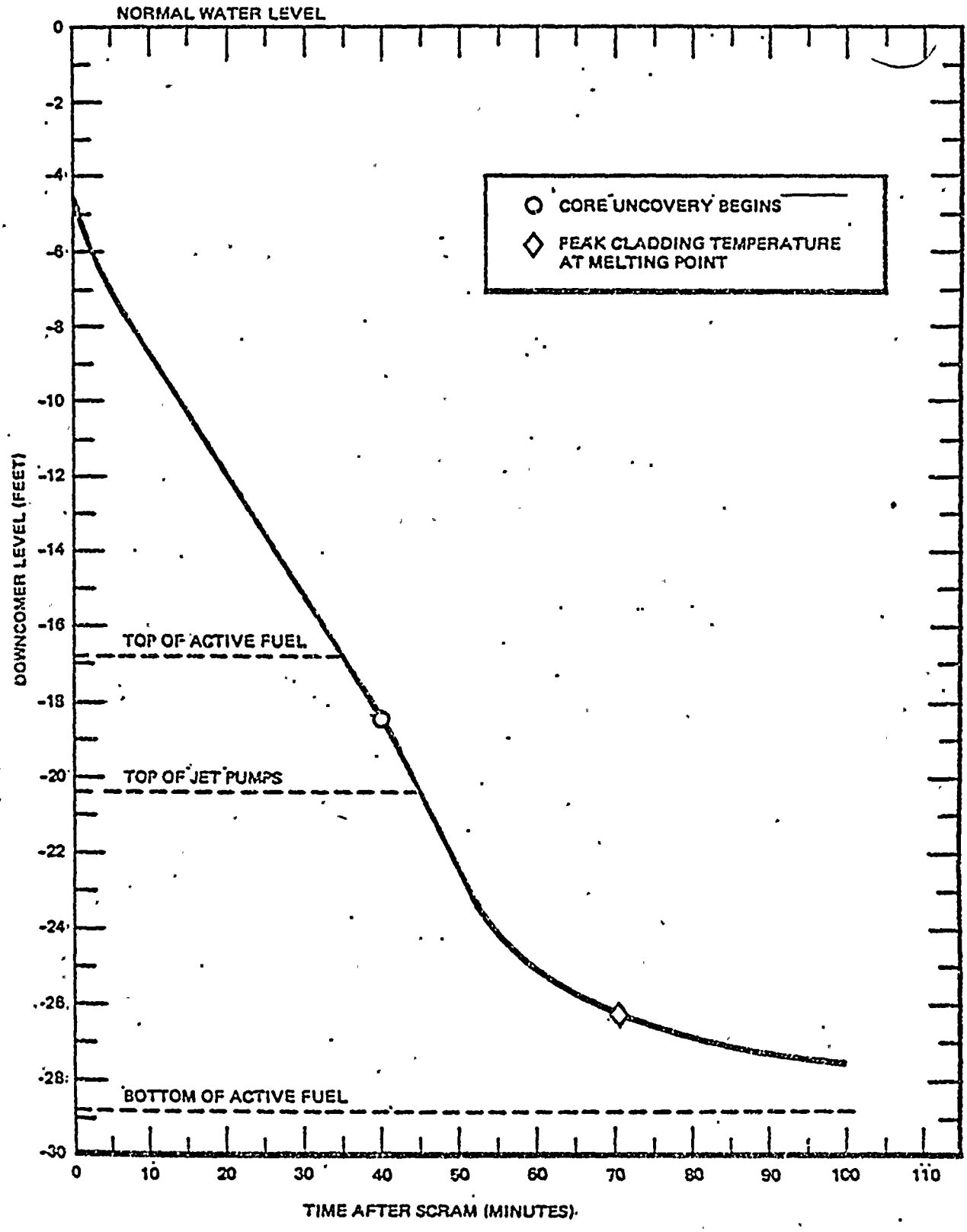
Conclusion

The BWROG study shows that knowledge of water level within the core is uniquely suitable and sufficient for the monitoring of the adequacy of core cooling under accident conditions. The existing water level measurement systems are highly reliable systems in providing information to the operator but that individual level measurement systems should be evaluated for possible improvements particularly with regard to loss of drywell cooling (which can produce flashing) and instrument line breaks.

Modifications can be made to reduce the probability of reactor water level instrumentation failure and thereby decrease its contribution to core melt from 13% to 3%. However the Susquehanna design already includes a significant portion of the improvements identified by the BWROG.

The addition of backup, diverse ICC detection devices is shown to have a very small additional contribution to overall risk reduction. Further, the safety priority analysis of these devices indicates a score in the lower end of the low priority range. Therefore, no additional instrumentation should be considered necessary for the detection of ICC because of its negligible contribution to plant safety.

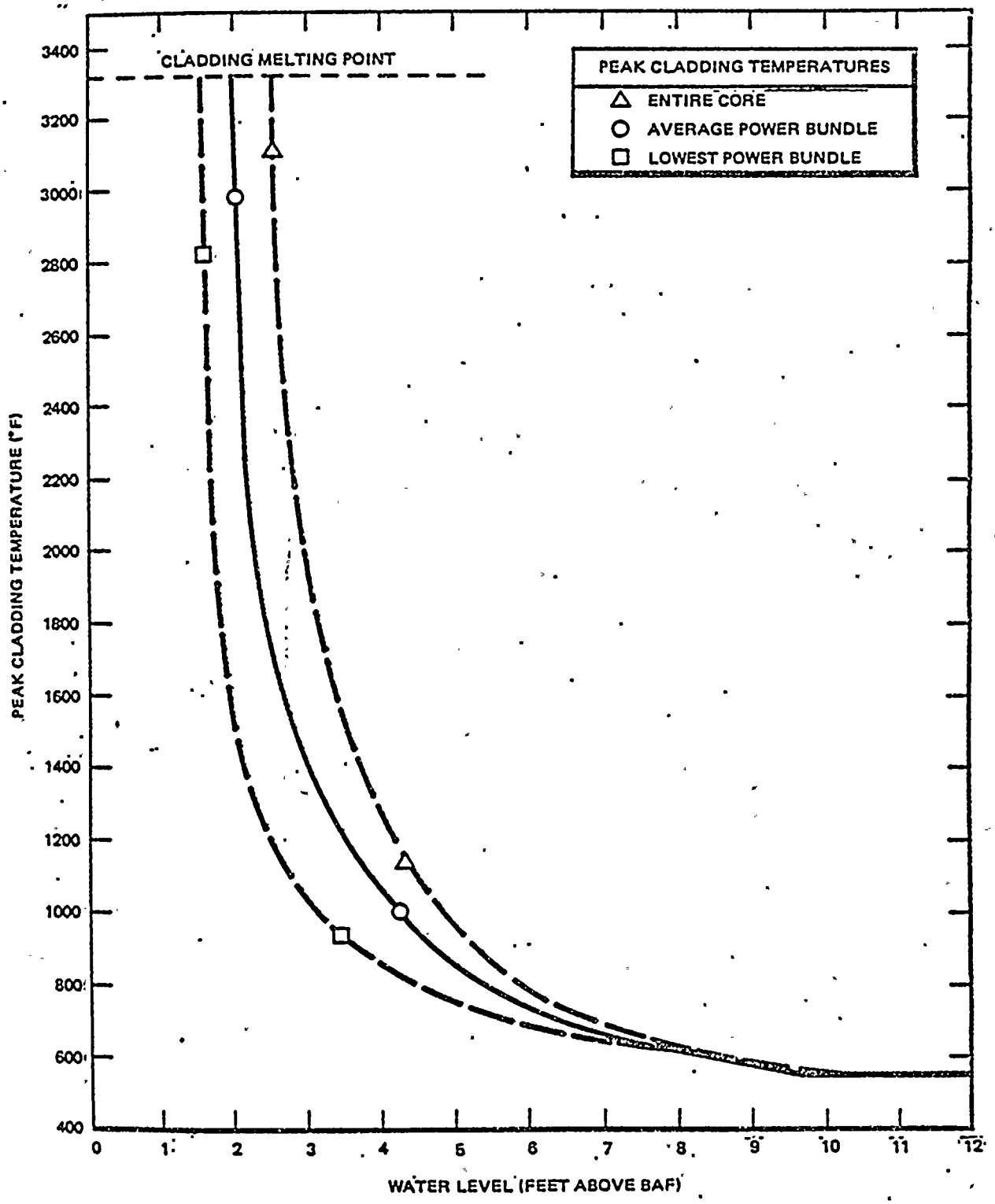
Excerpt f[illegible] SLI-8211, "Review of BWR Reacto[illegible] Vessel Water Level Measurement Systems"



Downcomer Water Level History

Figure 1

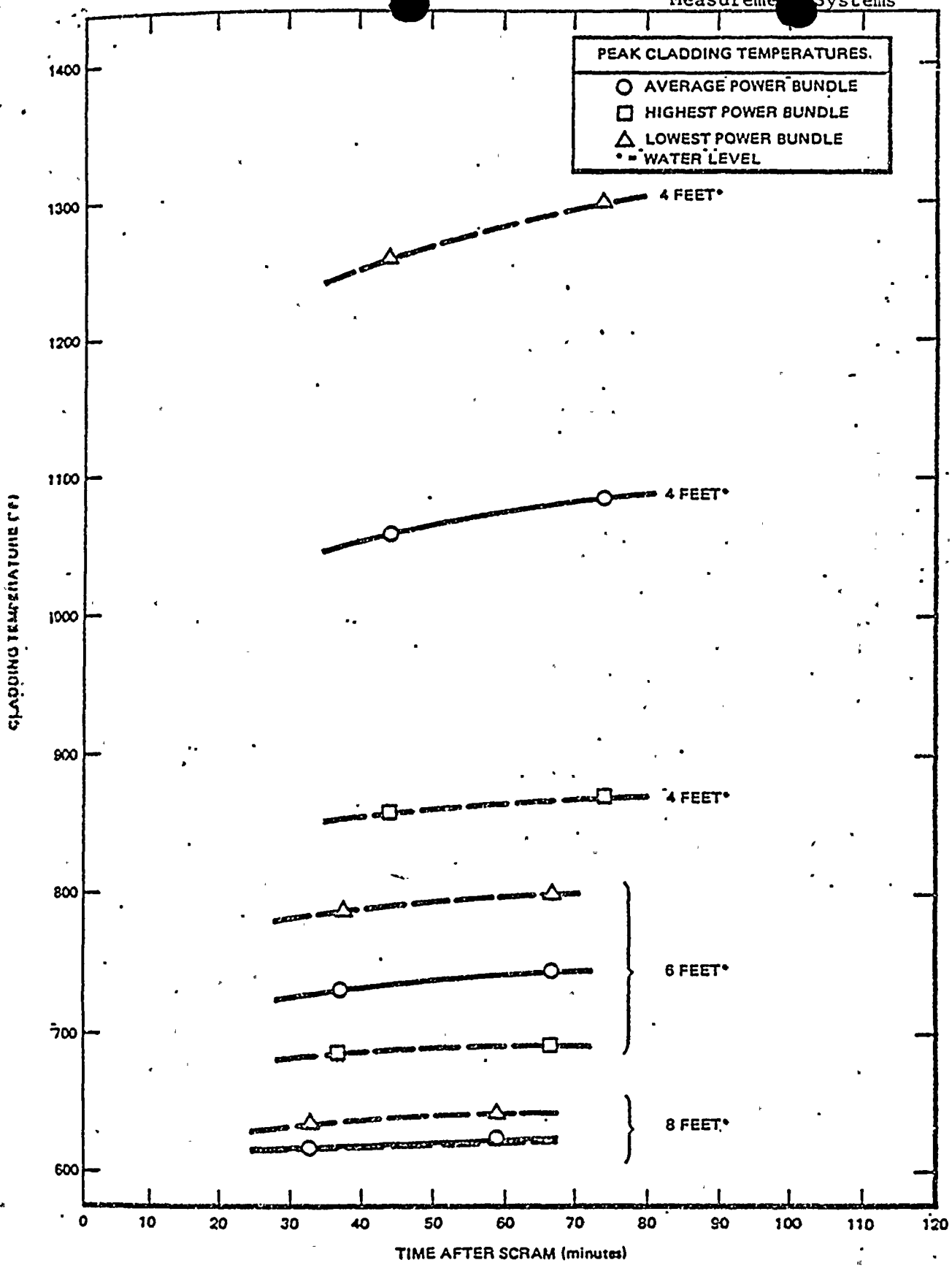
Excerpt from SLI-8211, "Review of BWR Reactor Vessel Water Level Measurement Systems"



Water Level As An Indicator of Core Overheating

Figure 2

Excerpt from SBI-8211, "Review of BWR Reactor Vessel Water Level Measurement Systems"



Cladding Temperature Sensitivity to Core Uncovery Time

Figure 3

Table 1

Possible ICC Detection Devices

Name of Device	Reference Number
Source Range Monitor	1
Intermediate Range Monitor	2
Local Power Range Monitor	3
Traveling Incore Probe	4
Gamma-Neutron Reaction Detector	5
Gamma Attenuation	6
Gamma Void Meter	7
Neutron Modulation Void Meter	8
Core Reactivity Detector	9
Fuel Plenum Tracer	10
Primary System Activity Meter	11
Incore Thermocouples	12
Heated Junction Thermocouples	13
Gamma Thermometers	14
Control Rod Drive Thermocouples	15
Sight Glass	16
Cerenkov Light Detector	17
Wave Guide	18
Vessel Weight	19
Vessel Vibrations	20
Floats	21
Conductivity Probe	22
Capacitance Probe	23

Table 1 , Continued

Name of Device	Reference Number
Sonic Reflection	24
Loose Parts Monitor	25
Microwave Probe	26
Mass Balance	27
Differential Expansion Integal Anemometer	28
Delta-P Bubbler	29
Self-Powered Neutron Detector	30
Resistance Temperature Detectors	31
Steam Dome Thermocouples	32
Liquid Level and Void Fraction Detector	33