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Thermal-Hydraulic Analysis of the Dual-Function Gamma Thermometer

J. O. Johnson
T. J. Burns

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THERMAL-HYDRAULIC ANALYSIS OF THE DUAL-FUNCTION GAMMA THERMOMETER

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NOTICE This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

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ACKNOWLEDGMENTS

The authors would like to acknowledge Katie Lawhorn and LaWanda Klobe who have patiently contributed their skills and talents toward the final preparation of this report.

ABSTRACT

A computational study was performed to further investigate the potential of a modified gamma thermometer as a monitor of both the localized power level and the adequacy of core cooling for a Pressurized Water Reactor.

The basic gamma thermometer has been proposed as an instrument for measuring the local heat generation rate within a reactor core. More specifically, the gamma thermometer can be viewed as a means for measuring the localized (e.g. within a particular assembly) power level. Thus, the GT can be envisioned as providing the localized data that defines the overall power distribution within the core--which can be summed to yield the global power level of the reactor.

The volumetric heat source within the thermometer was updated to include heat deposited by neutrons as well as heat deposited by decay gammas and neutron-induced gammas. Utilizing this source, along with a more detailed thermal-hydraulics model, a series of thermal-hydraulic calculations were performed to simulate certain reactor transients of interest (i.e., reactor scram, LOCA, etc.) in order to characterize the gamma thermometer response relative to power level monitoring and adequate core cooling indication.

The results of this study reaffirm the feasibility of utilizing the GT as a dual-function (power level and adequate core cooling capacity) measurement device, with each function accomplished virtually independent of the other. The study also indicates that there is a thermal-hydraulic regime for which the GT would no longer give easily interpretable signals. However, the pursuit of the GT as a viable nuclear instrument is still encouraged.

I. INTRODUCTION

A contributing factor to the severity of the Three-Mile Island-2 (TMI-2) accident was the lack of information available to the reactor operators regarding the thermal-hydraulic environment within the pressure vessel. The results of this study indicate that a modified version of a gamma thermometer (GT) has the potential for providing an indication of the localized thermal-hydraulic environment within a reactor core--specifically, an indication of the adequacy of core cooling capacity. In the case of the TMI-2 accident, the availability of such information might well have prompted the reactor operators to restore the Emergency Core Cooling System (ECCS), and consequently would have avoided the severe core damage that did occur.

The basic gamma thermometer was originally proposed as an instrument for the measurement of localized power generation within a reactor core.[1] It is significant to note that in the modified version this function is not compromised; i.e., the modified version is a true dual-function instrument. Thus, employment of the modified GT for PWRs [in place of the existing Self-Powered Neutron Detectors (SPNDs) currently used for power level monitoring] would result in a significant increase in available information regarding the state of the reactor core with no increase in the instrumentation required.

One clarification regarding the modified GT is in order, however. In the initial study of the dual-function GT,[2] the proposed instrument was analyzed as a power level indicator and a core coolant level monitor. If a well-defined coolant level exists in a reactor core, the modified GT is indeed capable of detecting such a situation. However, it is not clear whether, under accident conditions, such a well-defined level would exist. The basic thermal-hydraulic signal of the modified GT is related to the heat transfer coefficient of the reactor environment. Thus the modified GT actually provides a measure of the adequacy of the heat removal process in a localized region of the reactor irrespective of the medium by which this process is accomplished (i.e., water, steam, two-phase flow, etc.). Hence, regarding the modified GT as a fluid level monitor is too restrictive.

It is more accurate (but cumbersome) to describe the thermal-hydraulic signal of the modified GT as a measure of the adequacy of the localized cooling capacity of the thermal-hydraulic system.

The work reported here represents an extension of the initial GT study,[2] principally in the area of thermal-hydraulic modeling and analysis. As such, much of the analysis and information presented in the initial report has been omitted. Only the information required for continuity of this presentation is repeated here, and interested readers are referred to the initial document for further details.

II. BACKGROUND

2.0 Design Rationale

Recently consideration has been given to the possibility of using an old concept: the use of the gamma thermometer as a replacement for the SPNDs currently utilized for power level measurement in PWRs. As an example, the Tennessee Valley Authority (TVA) has initiated a $400,000 program to passively irradiate gamma thermometers in the Oak Ridge National Laboratory Research Reactor (ORNL-ORR) in order to assess the calibration versus irradiation characteristics.

This increased interest can be attributed primarily to the marked difference in the spatial variation of the thermal neutron and gamma fluxes within a fuel assembly. The reduced spatial variation of the gamma flux (as compared to the thermal neutron flux measured by the SPNDs) is anticipated to allow the accuracy of the power level measurement to be improved. Assuming that the improved accuracy in the measurement can be translated into a corresponding reduction in the uncertainty of the actual power generation rate, then the potential for increasing the operating limits (while maintaining the same nominal design limits) can be envisioned. Furthermore, by incorporating certain modifications into the gamma thermometer design, a dual-function instrument sensitive to both the power level and changes in the exterior heat transfer coefficient (and hence to the thermal-hydraulic environment within the core) is devised. It was anticipated and theoretically verified that the gamma thermometer, modified to include measurement of adequate core cooling potential, would provide a direct and unambiguous in-core indication of both the localized power level and the thermal-hydraulic environment within the reactor vessel.

2.1 Physical Description

The device itself consists of a hollow, cylindrical, stainless steel rod of a length equal to or greater than the height of the reactor core. At intervals along the rod, annuli of material are removed by machining. A series of differential thermocouples (TCs) are then located at each annulus location, with the TCs and associated electrical leads positioned in the center of the rod. Magnesium oxide is utilized as both packing and insulating material in the central cavity. Once assembled, zircalloy cladding is swaged onto the exterior in an inert atmosphere (typically argon). The resulting device is depicted in Fig. 1, and the associated thermocouple design is shown in Fig. 2. The assembled thermometer would then be typically inserted into the instrumentation guide tube of a fuel assembly.

2.2 Use as a Power Level Monitor

During operation of a nuclear reactor, the various neutron interaction processes (i.e., fission, capture, etc.) together with fission product decay produce gamma radiation as a by-product. The placement of a GT within a fuel assembly would allow some fraction of these gamma rays to interact with the stainless steel body of the proposed GT, depositing energy and thereby producing heat. The resulting heat is then transferred from the device to the coolant in which it is immersed. The presence of this volumetric heat source, coupled with the illustrated design, will produce a temperature distribution within the device itself. The incorporated thermocouples measure the magnitude of this temperature distribution at two locations (T_{HOT} and T_{COLD} in Fig. 1) within the standard device, with the difference between T_{HOT} and T_{COLD} being used to infer the localized heat generation rate (ΔV_1 in Fig. 2).

2.3 Use as an Adequate Core Cooling Monitor

Operation of the GT to monitor the adequacy of core cooling is based on the fact that the heat transfer coefficient on the exterior surface of the device depends on the state of the cooling medium within the fuel assembly. For example, the exterior heat transfer coefficient will undergo a dramatic change (roughly by a factor of 1,000) as the coolant changes phase from liquid to steam at constant pressure. The radial heat transfer characteristics at the "hot" and "cold" thermocouple locations are radically different under normal reactor operation (i.e., with the active region of the GT immersed in coolant). The radial flow at the "hot" thermocouple is sharply reduced due to the gas gap (which functions as an insulator).

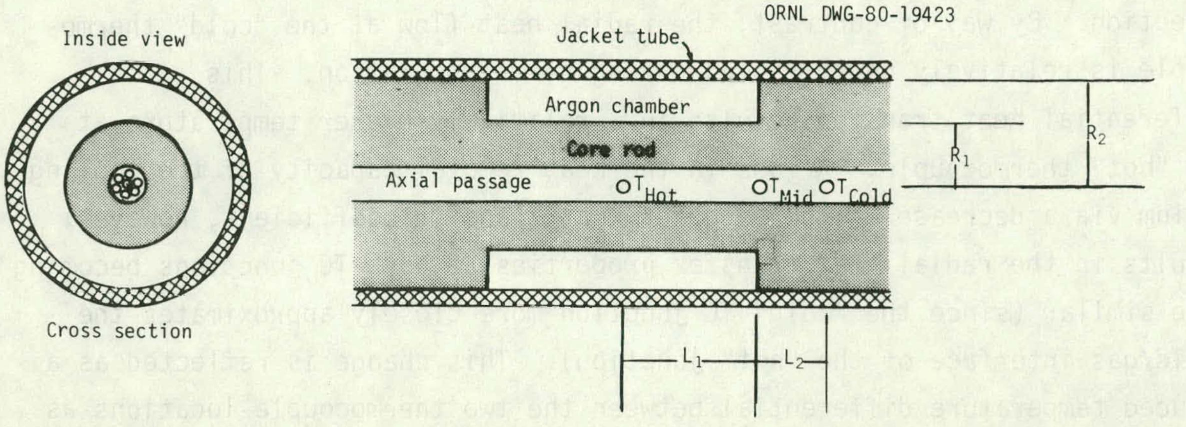

Fig. 1. Schematic of PWR gamma thermometer.

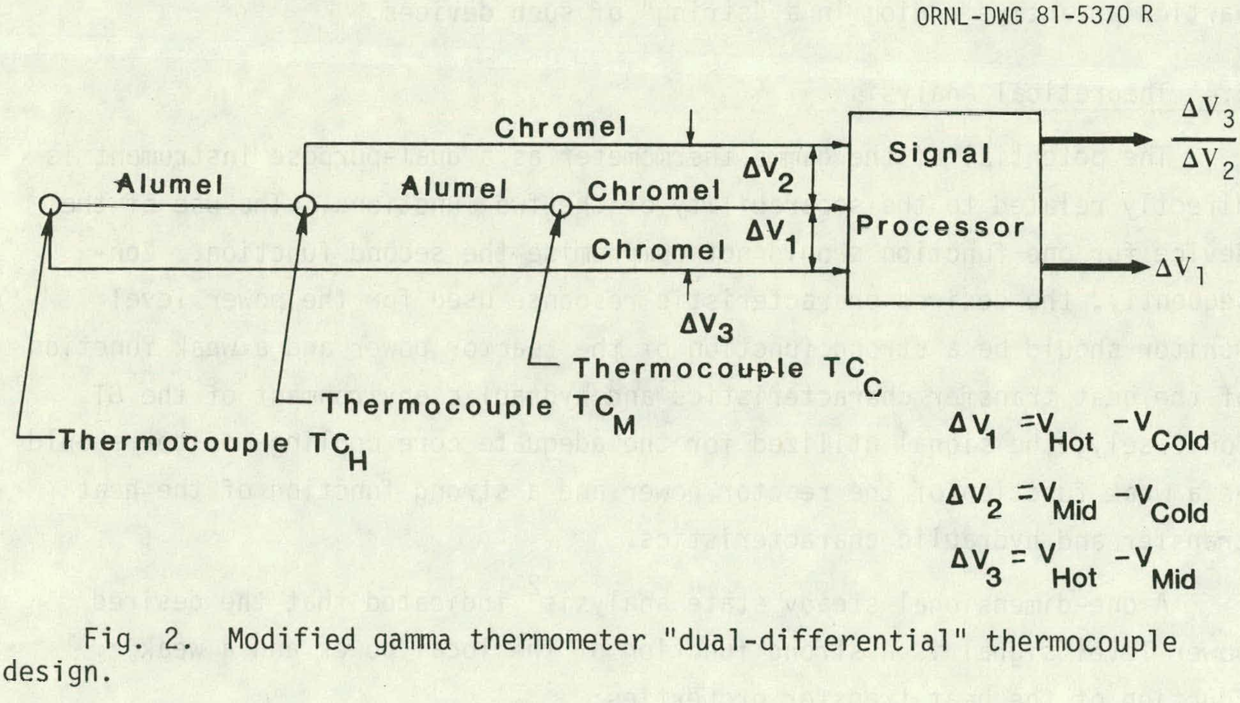

Fig. 2. Modified gamma thermometer "dual-differential" thermocouple design.

Therefore, the heat flow in this region is principally in the axial direction. By way of contrast, the radial heat flow at the "cold" thermocouple is relatively unrestricted during normal operation. This differential heat transfer results in a relatively higher temperature at the "hot" thermocouple. A loss in the heat removal capacity of the cooling medium via a decrease in the exterior heat transfer coefficient, however, results in the radial heat transfer properties at both TC junctions becoming more similar (since the "cold" TC junction more closely approximates the solid/gas interface of the "hot" junction). This change is reflected as a reduced temperature differential between the two thermocouple locations as well as higher absolute temperatures at both locations resulting from the lower overall heat removal capacity of the medium. The rationale for the use of the modified GT as a monitor of the adequacy of core cooling is that the shape of the temperature distribution (as measured by the thermocouples) is indicative of the degree of heat transfer similarity at the two junctions, and hence of the state of the coolant at the level of that particular active region in a "string" of such devices.

2.4 Theoretical Analysis

The potential of the gamma thermometer as a dual-purpose instrument is directly related to the separability of the two functions. The use of the device for one function should not compromise the second function. Consequently, the desired characteristic response used for the power level monitor should be a strong function of the reactor power and a weak function of the heat transfer characteristics and hydraulic environment of the GT. Conversely, the signal utilized for the adequate core cooling monitor should be a weak function of the reactor power and a strong function of the heat transfer and hydraulic characteristics.

A one-dimensional steady state analysis[2] indicated that the desired power level signal is a strong function of the local power and a weak function of the heat transfer properties:

$$(T_{HOT} - T_{COLD}) = \frac{qL_1^2}{2k} + \frac{qL_1R_1^2}{kmR_2^2}\left[\frac{(1 + e^{-2mL_2}) - 2e^{-mL_2}}{(1 - e^{-2mL_2})}\right] \qquad (1)$$

where q is the volumetric heat deposition rate (related to the local power) and $m^2 = \frac{hp}{kA}$. The dimensions are given in Fig. 1.

By utilizing an additional temperature measurement at a point between the "hot" and "cold" junctions (denoted as T_{MID} in Figs. 1 and 2), a response that is insensitive to the power level but strongly dependent on the surface heat transfer coefficient (via m) can be constructed:

$$\frac{(T_{HOT} - T_{MID})}{(T_{MID} - T_{COLD})} = \frac{\frac{mL_1R_2^2}{2R_1^2}(1 - e^{-2mL_2})}{[(1 + e^{-2mL_2}) - 2e^{-mL_2}]} \qquad (2)$$

Note that Eq. (2) does not depend explicitly on q, the volumetric heat deposition rate. An additional favorable characteristic of this design is that the final thermal-hydraulic signal requires no additional information other than that available from the GT thermocouple measurements themselves (i.e., the instrument is self-contained).

Physically, the two signals will be measured as voltage drops across the differential thermocouples placed in series (see Fig. 2). One relatively straightforward manner of obtaining the requisite information is to incorporate the "dual differential" thermocouple design (depicted in Fig. 2), measuring $T_{HOT} - T_{MID}$ (ΔV_3 in Fig. 2) with one differential TC and $T_{MID} - T_{COLD}$ (ΔV_2 in Fig. 2) with a second differential TC. The ratio of the two signals would yield the adequate core cooling monitor response, and the sum of the two signals would yield the power level monitor response. Alternatively, two differential thermocouples (four leads versus three) could be used for the same measurements.

2.5 Radiation Transport Analysis

In the initial study,[2] a detailed radiation transport analysis was performed using the DOT-IV[3] discrete ordinates transport code to calculate and characterize the volumetric energy deposition rate within the GT [i.e., to determine q in Eq. (1)]. The characterization included the source of the particles involved (i.e., geometrically within the reactor), the origin of the particles (i.e., neutron-induced reactions or fission product decay), and also the manner in which the particles actually reach the detector. For the present analysis, the additional contribution of neutron heating to the total energy deposition rate within the thermometer was determined and is depicted in Fig. 3. This result, coupled with the contribution from fission product decay gammas (Fig. 4) and neutron-induced gammas (Fig. 5), yields the total volumetric energy deposition rate shown in Fig. 6. The percentage breakdown of this rate is 4.3% due to the neutron heating, 20.8% due to fission product decay, and 74.9% due to neutron-induced reactions (with 47.1% of the neutron-induced reactions, or 35.3% of the total signal, being attributable to fission). Inclusion of the neutron heating in the total energy does not affect the spatial distribution of the total energy deposition rate. The results continue to indicate a relatively flat spatial distribution within the gamma thermometer itself, with the maximum spatial deviation from centerline value being approximately 6%. Thus, much of the previous characterization of the power level function remains valid and will not be repeated here.

radius (cm)	0.0–0.0853	0.0853–0.1706	0.1706–0.2559	0.2559–0.3412	0.3412–0.4265
0.3412–0.4265	0.2987	0.3124	0.3331	0.3535	0.3652
0.2559–0.3412	0.3036	0.3145	0.3305	0.3453	0.3535
0.1706–0.2559	0.3191	0.3223	0.3268	0.3305	0.3331
0.0853–0.1706	0.3387	0.3321	0.3223	0.3145	0.3124
0.0–0.0853	0.3533	0.3387	0.3191	0.3036	0.2987

radius (cm)

Fig. 3. Volumetric energy deposition rate distribution in the gamma thermometer due to neutron heating (W/cm^3).

radius (cm)	0.0–0.0853	0.0853–0.1706	0.1706–0.2559	0.2559–0.3412	0.3412–0.4265
0.3412–0.4265	1.5921	1.6095	1.6359	1.6694	1.7072
0.2559–0.3412	1.5646	1.5794	1.6049	1.6356	1.6694
0.1706–0.2559	1.5621	1.5691	1.5858	1.6049	1.6359
0.0853–0.1706	1.5729	1.5716	1.5691	1.5794	1.6095
0.0–0.0853	1.5862	1.5729	1.5621	1.5646	1.5921

radius (cm)

Fig. 4. Volumetric energy deposition rate distribution in the gamma thermometer due to fission product decay source (W/cm^3).

radius (cm)	0.0–0.0853	0.0853–0.1706	0.1706–0.2559	0.2559–0.3412	0.3412–0.4265
0.3412–0.4265	5.5428	5.5846	5.6259	5.6559	5.6487
0.2559–0.3412	5.6346	5.6654	5.6905	5.6965	5.6559
0.1706–0.2559	5.7254	5.7287	5.7215	5.6905	5.6259
0.0853–0.1706	5.8020	5.7766	5.7287	5.6654	5.5846
0.0–0.0853	5.8646	5.8020	5.7254	5.6346	5.5428

radius (cm)

Fig. 5. Volumetric energy deposition rate distribution in the gamma thermometer due to all neutron-induced reactions (W/cm^3).

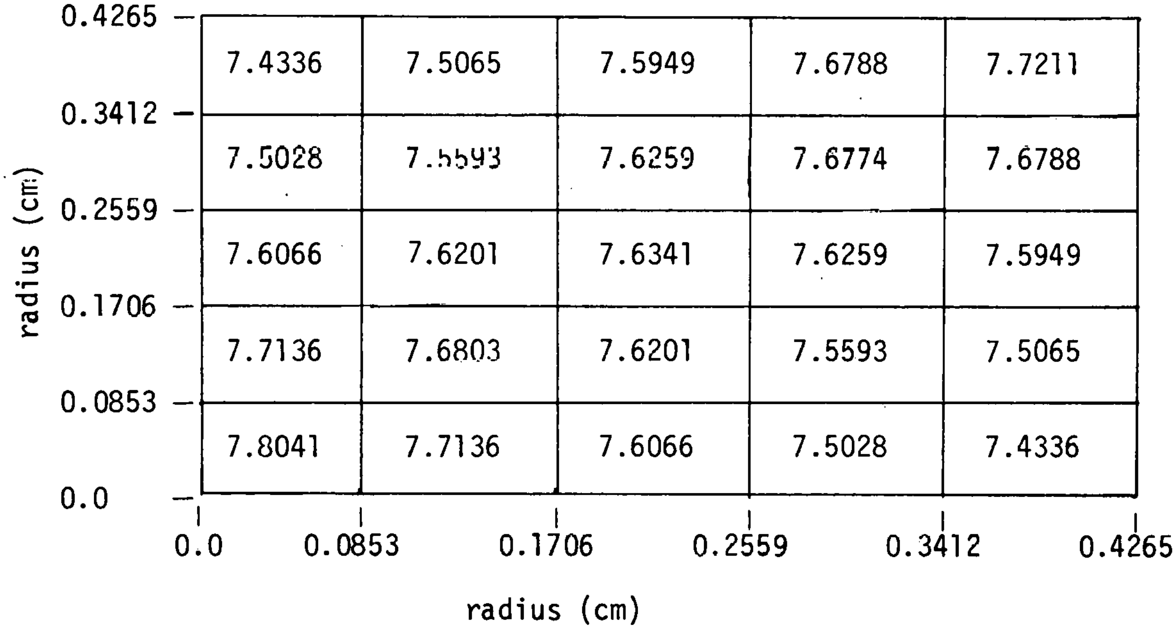

Fig. 6. Total volumetric energy deposition rate distribution in the gamma thermometer due to all sources (decay + neutron induced + neutron heating) [W/cm^3].

III. THERMAL HYDRAULIC ANALYSIS

Although the previous thermal-hydraulic analysis indicated the potential of the gamma thermometer as a means for addressing the adequacy of the cooling process, certain questions remained concerning the behavior of the device under accident conditions. To address these questions, particularly those related to the effect of coolant-related parameters on the gamma thermometer, a new prototypic calculational model of the GT (depicted in Fig. 7) was constructed. This model differs from the previous thermal-hydraulic model in four significant aspects: (1) revision of the physical dimensions to incorporate the results of a simple design sensitivity analysis, (2) extension of the model to address possible asymmetry in the axial direction, (3) an increase in the modeling detail from 100 to 625 nodes to obtain a finer (and hence more accurate) temperature distribution, and (4) incorporation of a more precise representation of the boundary conditions for the argon gap.

The model was utilized in the transient heat conduction code HEATING-5[4] to calculate both the spatial and time-dependent behavior of the gamma thermometer temperature distribution using cooling parameters typical of the environment within a PWR fuel assembly. The analysis incorporated the total volumetric heat source (Fig. 6) obtained via the radiation transport calculations together with temperature-dependent material properties from the Nuclear Systems Materials Handbook.[5]

The analysis considered the behavior of the GT signals (see Fig. 2) during normal reactor operating conditions as well as during and subsequent to various reactor transients. The transients analyzed were: (1) a reactor scram, modeled as an instantaneous termination of the gamma source attributable to neutron-induced reactions; (2) an instantaneous loss of coolant accident (LOCA), modeled as an instantaneous change in the external heat transfer coefficient from approximately 30,000 W/m^2 °C (which represents normal reactor core conditions) to approximately 1,800 W/m^2 °C (which represents saturated steam at 15.5 MPA and 315°C); (3) a combination of a reactor scram and LOCA at 15.5 MPA, (4) a large pipe break LOCA, modeled as an instantaneous change in the external heat

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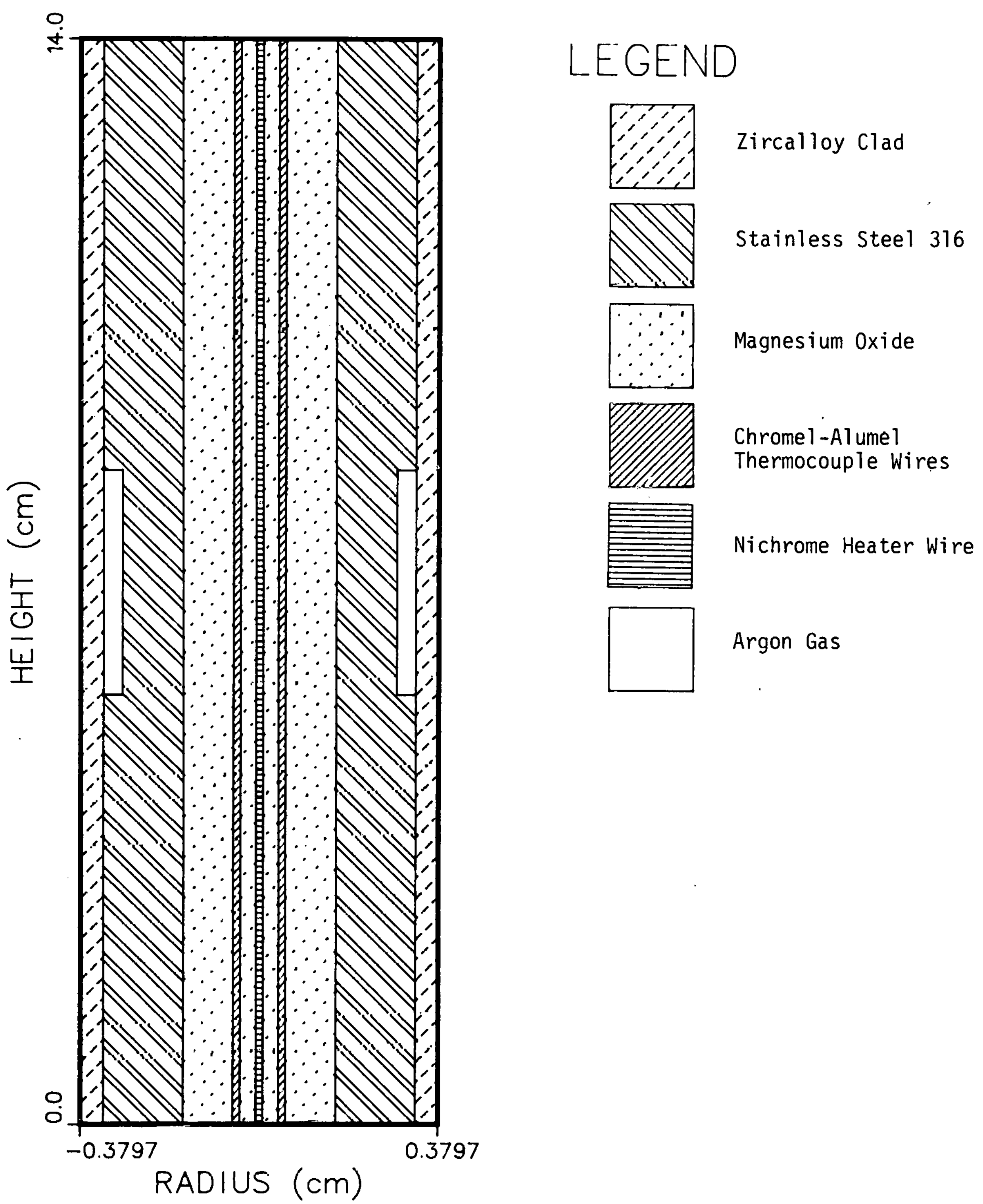

Fig. 7. Calculational model of the gamma thermometer used for thermal-hydraulics analysis.

transfer coefficient from approximately 30,000 W/m^2 °C to 50 W/m^2 °C (which represents saturated steam at 0.1 MPA and 315°C); (5) a combination of a reactor scram and LOCA at 0.1 MPA; and (6) a partial LOCA, modeled as an instantaneous drop in the coolant level which uncovers the upper half of the thermometer (i.e., from the top of the GT to the center of the argon gap in Fig. 7) to a saturated steam environment at 15.5 MPA and 315°C.

Two characteristic parameters are considered appropriate regarding the applicability of the GT as a power level monitor: the calibration of the device with respect to the Local Heat Generation Rate (LHGR), and the time constant of the instrument itself. The calibration of the device with respect to the LHGR can be expressed as

$$LHGR = \alpha \cdot (T_{HOT} - T_{COLD}) + \beta \tag{3}$$

where α is the proportionality constant relating the local heat generation rate to the temperature differential and β is an adjustment factor, required since the ratio of fission product LHGR to total LHGR is not identical to the ratio of fission product GT signal to total signal.

The previous study[2] indicated values for α and β of 14.96 W/cm °C and -55.19 W/cm respectively. As a result of the changes in the thermal-hydraulics model, specifically the lengthening of the argon gap, the slope (i.e., α) of the calibration curve depicted in Fig. 8 changed considerably. The current design produces values for α and β of 8.88 W/cm °C and -56.63 W/cm respectively. The significance of this result is that it indicates the large sensitivity of the power level signal to the physical design and fabrication of the instrument. It also apparently indicates that each detector must be calibrated separately.

The second characteristic of the gamma thermometer that influences its acceptability as a power level monitor is the time constant of the instrument itself. The results of this study indicate a thermocouple response time of approximately 0.0856 °C/s (as compared to the previous value of 0.0508 °C/s). This change is also attributable to the alterations in the thermal-hydraulics model, particularly the dimensional changes.

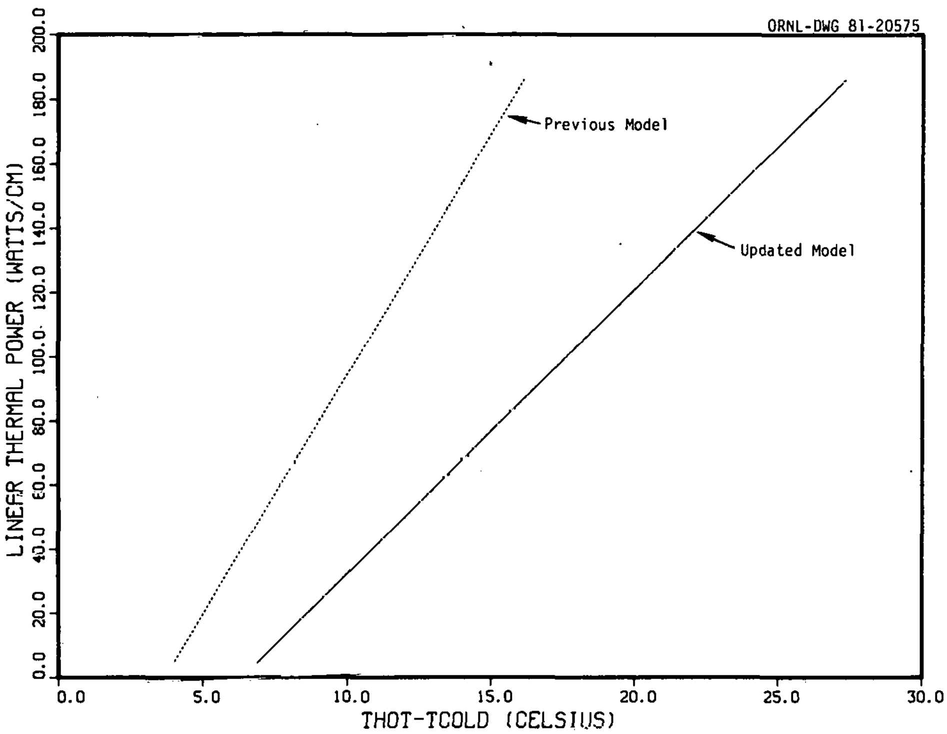

Fig. 8. Comparison of calibration curves for the power level indicator (33,800 MWD/MTHM burnup).

A factor of four change in the GT power level indicator signal as a result of an instantaneous reactor scram had been determined previously. Further, the time dependence of the power level signal subsequent to an instantaneous LOCA at 15.5 MPA (after allowing for an initial transient effect) returned to its initial value. The same transients were calculated using the updated thermal-hydraulics model of the GT. Figure 9, depicting the power level signal subsequent to a reactor scram, and Fig. 10, depicting the power level signal subsequent to an instantaneous LOCA at 15.5 MPA, provide additional confirmation of the initial results, maintaining approximately a factor of four change in the power level signal for a reactor scram. Similarly, the power level indicator signal for a combination reactor scram and LOCA at 15.5 MPA (Fig. 11) and at 0.1 MPA (Fig. 12) show the appropriate response (i.e., a factor of four drop consistent with the strong dependence on the reactor power level). In comparing Figs. 9, 11, and 12, the results show no discernable differences.

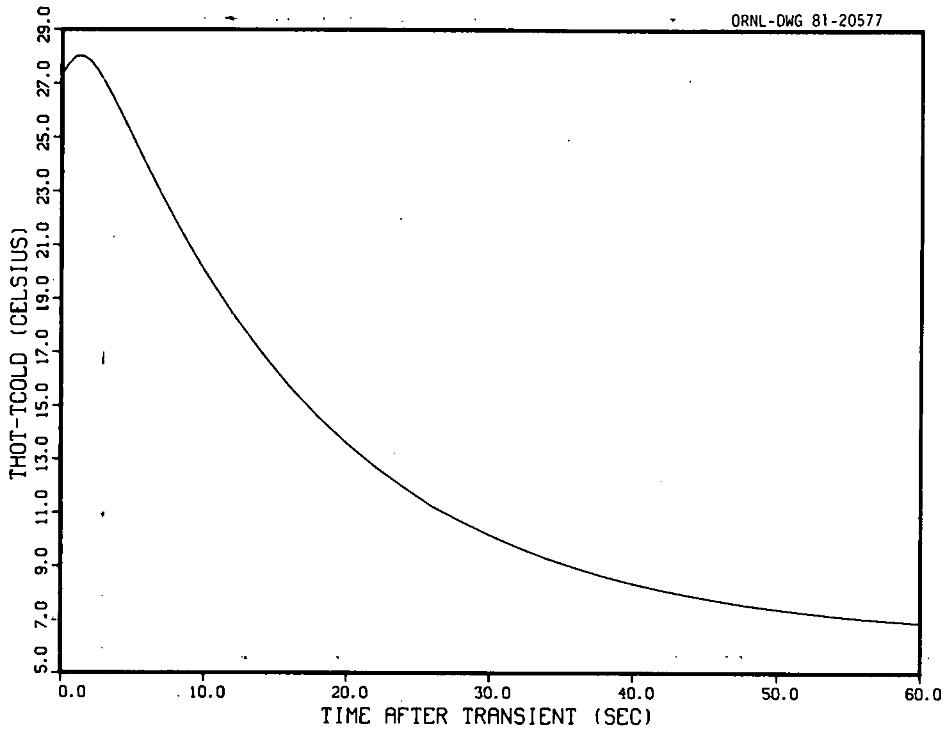

Fig. 9. Power level indicator response subsequent to an instantaneous reactor scram.

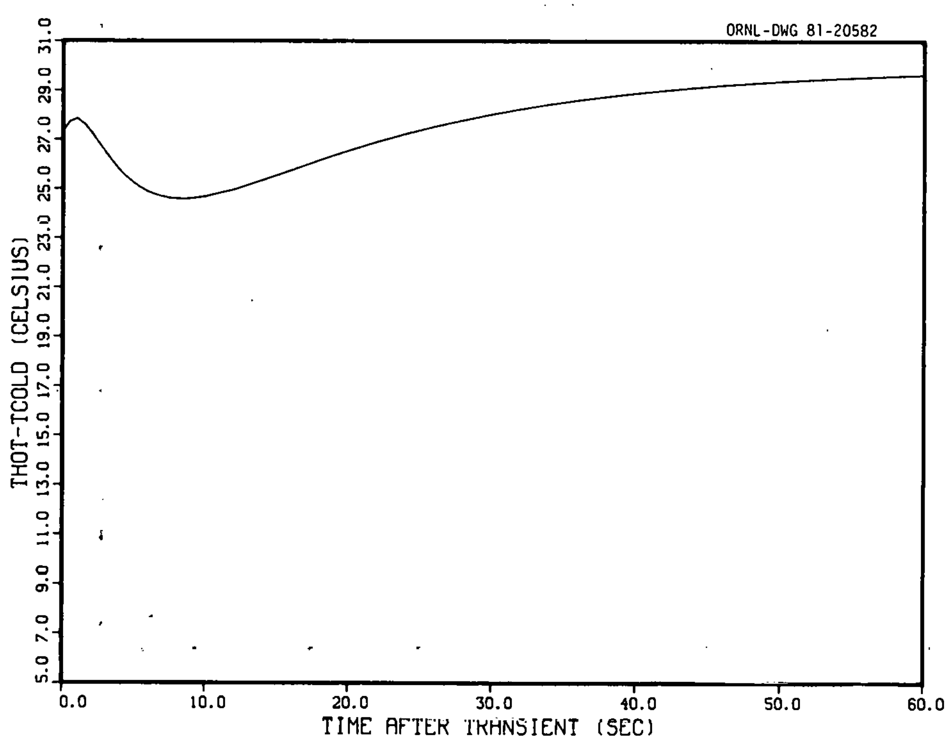

Fig. 10. Power level indicator response subsequent to an instantaneous loss of coolant accident (LOCA) at 15.5 MPA and 315°C.

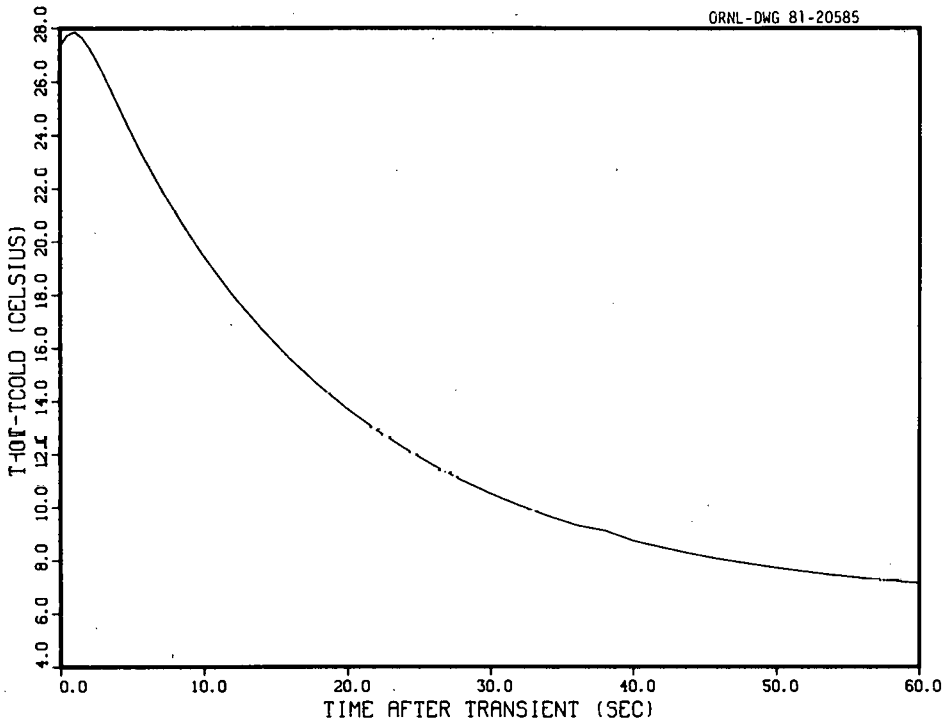

Fig. 11. Power level indicator response subsequent to a combination reactor scram and LOCA at 15.5 MPA and 315°C.

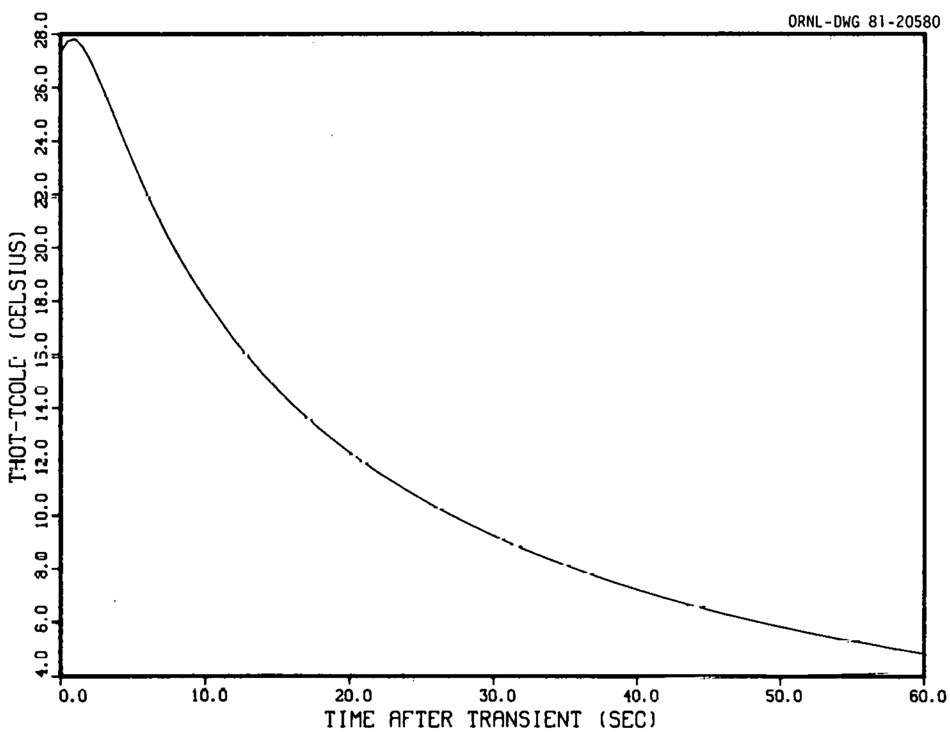

Fig. 12. Power level indicator response subsequent to a combination reactor scram and LOCA at 0.1 MPA and 315°C.

Furthermore, the power level indicator signal for the partial LOCA (at 15.5 MPA) traced the time response curve for the complete LOCA at 15.5 MPA (Fig. 10), so that there was no discernable difference between the two curves. These results substantially uphold the initial contention that the power level indicator response is virtually independent of the thermal-hydraulic environment within the fuel assembly. As an extreme case, the large pipe break LOCA was modeled. This calculation is noteworthy in that the results indicate a significant limitation inherent in the gamma thermometer. The power level indicator response for an instantaneous LOCA at 0.1 MPA (Fig. 13) exhibits a factor of four drop in the signal. This is a direct contradiction to all previous results, in which essentially no change in the power level signal was observed. The analysis of this result indicates that there are thermal-hydraulic regimes for which the GT no longer produces easily interpretable signals. Preliminary analysis of this calculation indicates that this phenomenon occurs when the radiative heat transfer mechanism becomes the dominant mode of heat removal from the GT. Consequently, the heat transfer process is no longer linear with the change in temperature (as is the case with the forced convective heat transfer coefficient). It is also noteworthy that this effect did not appear for the combined reactor scram and LOCA transient at 0.1 MPA (Fig. 12) due to the concurrent decrease in the power level. For this case, the accompanying reduction in the power level permitted operation of the gamma thermometer to remain in a temperature range where the radiative heat transfer mechanism was not the dominant mode. In summary, the overall results indicate that the gamma thermometer's operation as a power level monitor is insensitive to the thermal-hydraulic environment (except for the case of the most severe accident, a large pipe break LOCA with no reactor scram), and will yield a signal response proportional to the localized power.

The calculations cited above also confirm the thermal-hydraulic response of the GT [i.e., the adequate core cooling monitor (T_{HOT} - T_{MID}/T_{MID} - T_{COLD})] to be a strong function of the thermal-hydraulic environment (specifically the exterior heat transfer coefficient) and to be virtually independent of the reactor power. This result is illustrated

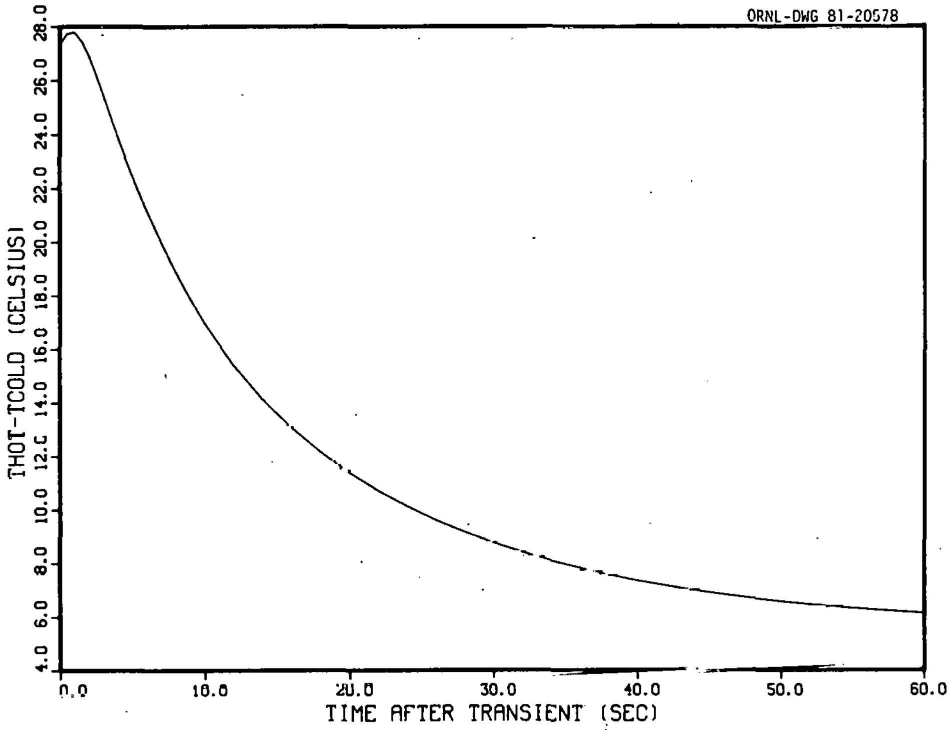

Fig. 13. Power level indicator response subsequent to an instantaneous LOCA at 0.1 MPA and 315°C.

in Figs. 14 and 15, which represent the adequate core cooling monitor (ACCM) response to an instantaneous LOCA at 15.5 MPA and to a combination reactor scram and LOCA at 15.5 MPA respectively. In each case, the results show a factor of two decrease in the signal immediately following the transient. Again it should be noted that for the partial LOCA at 15.5 MPA, the ACCM response traced the response for the complete LOCA (Fig. 14) and is therefore not displayed separately. The insensitivity of the ACCM signal to the reactor power level is depicted by Fig. 16, which shows the gamma thermometer signal during and subsequent to a reactor scram. The noise apparent in the signal is the result of roundoff errors in the calculation and is well within the error band for a differential thermocouple. The results of the ACCM response during and subsequent to a LOCA at 0.1 MPA and a combination reactor scram and LOCA at 0.1 MPA are presented as Figures 17 and 18 respectively. These results would appear to yield the best response of any of the transients (i.e., a factor of 14 drop in the ACCM signal instead of a factor of 2). However, as pointed out above,

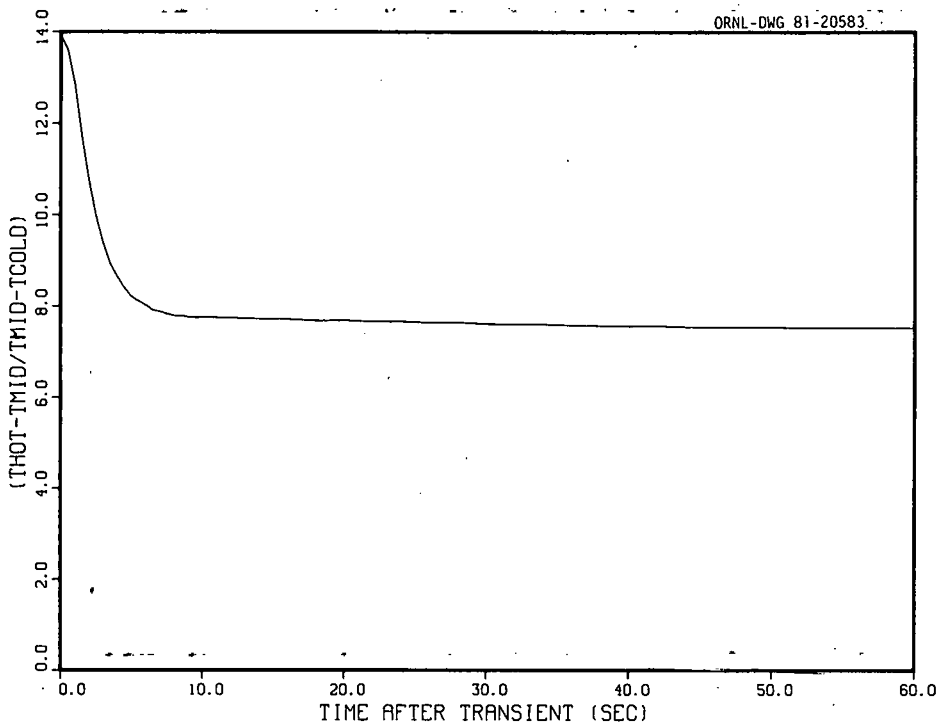

Fig. 14. Adequate core cooling monitor response to an instantaneous LOCA at 15.5 MPA and 315°C.

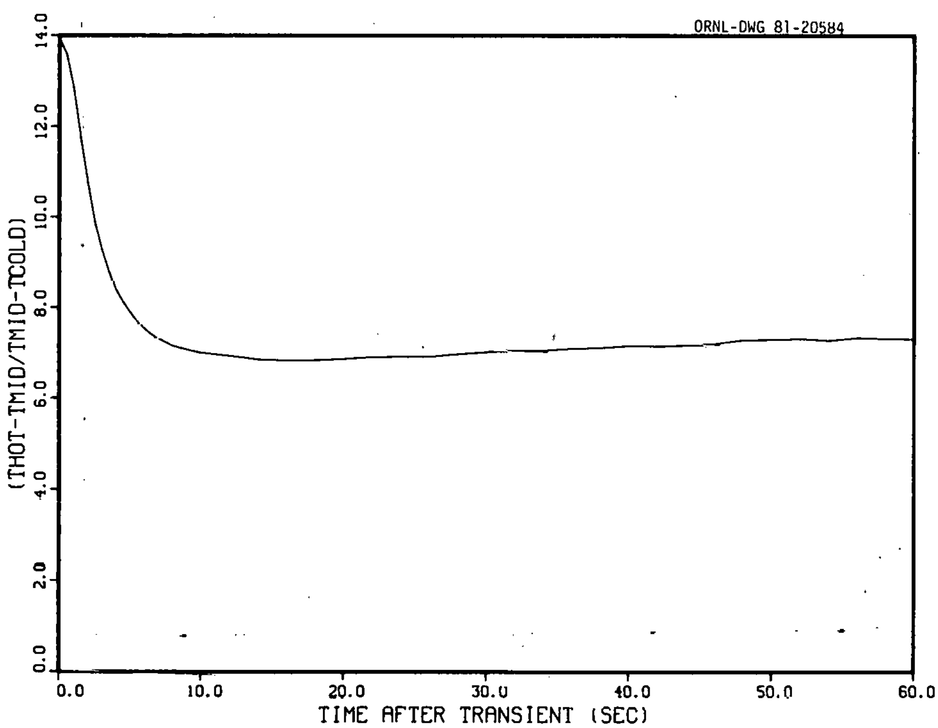

Fig. 15. Adequate core cooling monitor response to a combination reactor scram and LOCA at 15.5 MPA and 315°C.

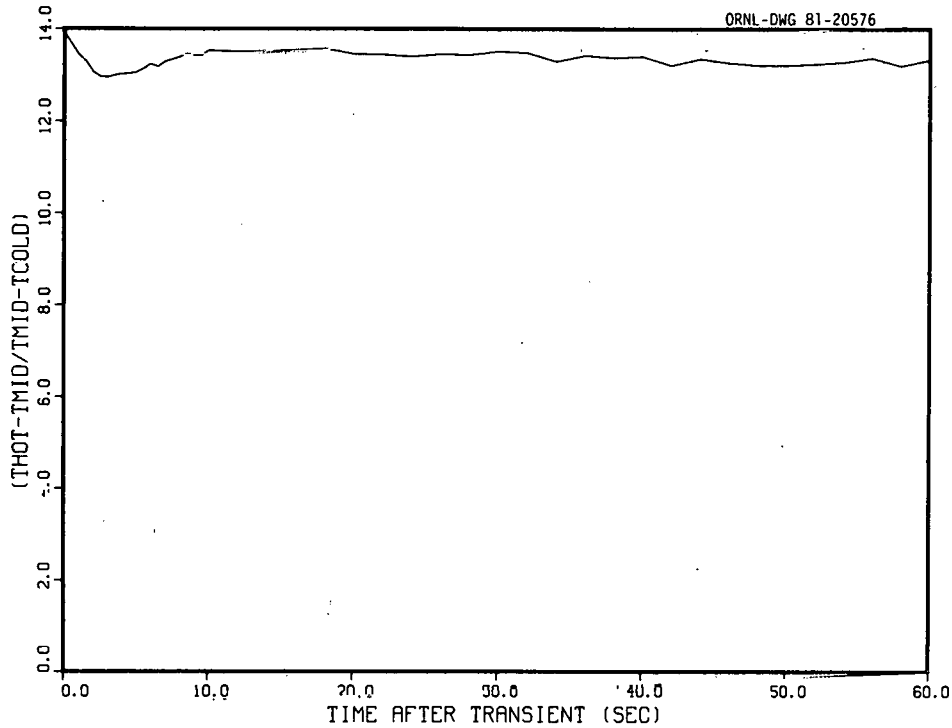

Fig. 16. Adequate core cooling monitor response to an instantaneous reactor scram.

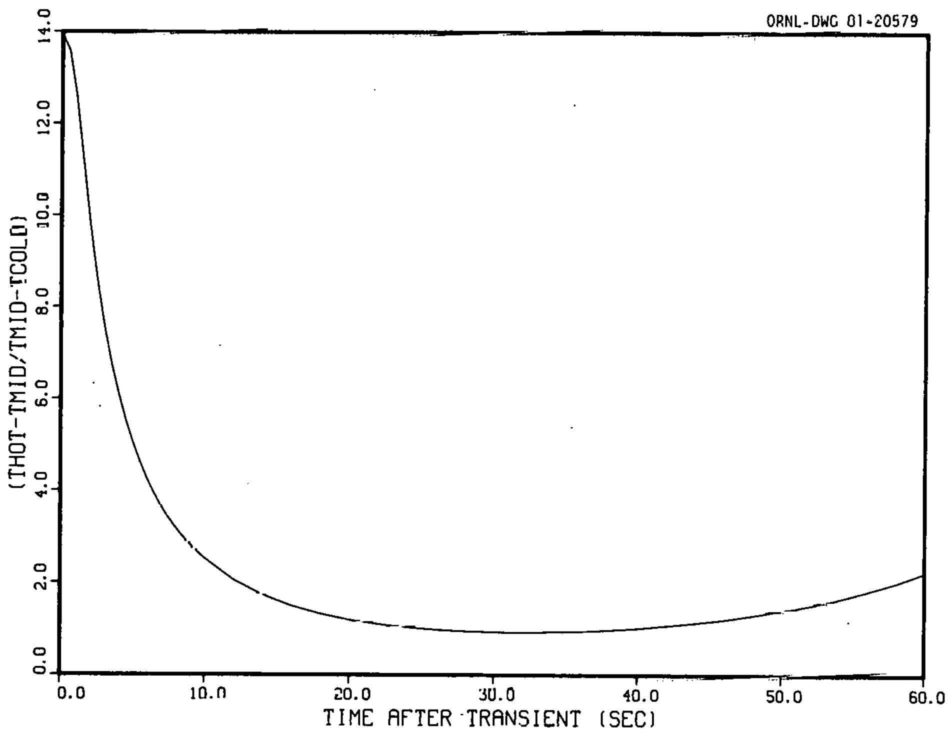

Fig. 17. Adequate core cooling monitor response to an instantaneous LOCA at 0.1 MPA and 315°C.

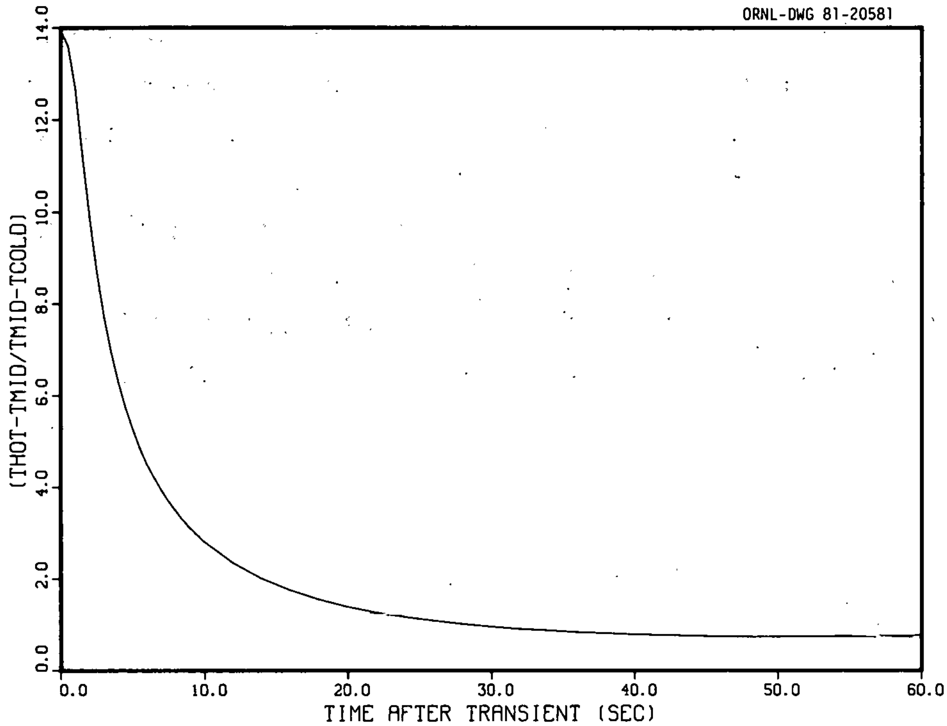

Fig. 18. Adequate core cooling monitor response to a combination reactor scram and LOCA at 0.1 MPA and 315°C.

the thermometer is producing a signal fundamentally different from the previous cases. As a result of the changes in the dominant heat transfer mode, the two effects (scram and LOCA) are no longer independent and separable. For the LOCA at 0.1 MPA (Fig. 17), the results indicate signal increase after reaching a minimum around 35.0 seconds. Although not shown, the signal eventually returns to its original value as though no transient had occurred.

The case of the combination reactor scram and LOCA at 0.1 MPA is, however, significantly better regarding interpretability. Even though the signal does reach a minimum, there is only a slight increase following the minimum (to a constant value of 1.40). Although the response of the ACCM is not as definitive for this case due to the increased radiative heat transfer, the sharp initial drop in the ACCM reading will still indicate the inadequacy of the cooling process to the reactor operator, prompting corrective measures.

The results presented in this section indicate that the time-dependent power level indication is a strong function of the reactor power (Fig. 9), yet is insensitive to the thermal-hydraulic environment (Fig. 10). Similarly, the ACCM response was demonstrated to be a strong function of the thermal-hydraulic environment (Fig. 14) but insensitive to the reactor power (Fig. 16). Finally, the results of the combined reactor scram and LOCA substantiate the results obtained in the previous study, further strengthening the claim that the gamma thermometer can be utilized as a dual-purpose measurement device (i.e., power level monitor and ACCM) without compromising its effectiveness for either function.

IV. CONCLUSIONS AND RECOMMENDATIONS

The initial contention that the gamma thermometer can be used for two important but disparate functions (power level and adequate core cooling) has been strengthened by the more detailed calculations described in this report. Moreover, the results presented here confirm that to a large extent each function can be isolated from the other. The signal utilized to indicate the power level is proportional to the LHGR and is insensitive to the thermal-hydraulic environment. Conversely, the signal employed as a monitor of adequate core cooling is responsive to changes in the thermal-hydraulic environment but is not particularly sensitive to changes in reactor power. Thus, it should be possible to infer the state of the reactor even if both the power level and thermal-hydraulic environment change simultaneously, as would occur in the case of a reactor scram initiated by a small break LOCA.

This report also provides further clarification of the initial contention of defining the modified GT as a possible coolant level detector. If a well-defined coolant level exists in a reactor core, the modified GT has been shown to be capable of detecting such a situation. However, because it is not clear whether a well-defined level exists, especially under accident conditions, and since the GT signal is based on the exterior heat transfer coefficient, the GT actually provides a measure of the adequacy of the heat removal process, irrespective of the medium by which this process is accomplished. Therefore, regarding the GT as only a coolant level monitor is restrictive, and a more appropriate definition of its function would be measurement of the adequacy of the localized cooling capacity of the thermal-hydraulic system. Utilization of the gamma thermometer in this mode (i.e., ACCM) indicates that the level of the coolant is of secondary importance so long as the operators know that the volume of coolant present can sufficiently remove the core heat and maintain the integrity of the fuel assemblies.

Extending the analyses of the gamma thermometer response to include the extreme accident scenario of a large pipe break LOCA (both coolant and pressure loss) indicated that the thermometer has a limit to which the signal would yield easily interpretable results. The analysis of the data showed the breakdown in the gamma thermometer signal to be due to a change in the primary mode of heat transfer from convection (both forced and natural) to radiation.

Although the results obtained in this study support the results of the previous study, this study must still be regarded as preliminary in nature--comprising only the theoretical characterization of the device. In particular, the adequacy of the many approximations and assumptions necessary to perform this study must be validated by experiment. In conjunction with the recommendations presented in the previous report, additional areas requiring further study or more detailed analysis are:

1. Determination of the exact point at which the gamma thermometer signal is no longer interpretable (i.e., where the heat transfer mode is primarily through thermal radiation, or where noise obscures legitimate readings.)

2. Determination of the effect of two-phase flow on the GT signals; in particular, the effect of a thin film of coolant on the exterior of the GT.

3. Determination of the adequacy of the gamma thermometer to model the thermal-hydraulic environment experienced by a fuel pin during a LOCA.

4. Analyze the gamma thermometer response to a real accident scenario such as the TMI-2 accident, and determine whether the device would have given the reactor operators indication of inadequate core cooling.

5. Fabricate an experimental prototype to be tested in a pressurized facility for the purpose of validating the theoretical results experimentally.

REFERENCES

1. R. H. Leyse and R. D. Smith, "Gamma Thermometer Development for Light Water Reactors," *IEEE Transaction on Nuclear Science,* NS-26, 934-942 (1979).

2. T. J. Burns and J. O. Johnson, "Theoretical Characterization of a Dual Purpose Gamma Thermometer," ORNL/TM-7567 (1980).

3. W. A. Rhoades, D. B. Simpson, R. L. Childs, and W. W. Engle, "The DOT-IV Two-Dimensional, Discrete-Ordinates Transport Code with Space-Dependent Mesh and Quadrature," ORNL/TM-6529 (1978).

4. W. D. Turner, D. C. Elrod, and I. I. Simon-Tov, "HEATING-5 — An IBM 360 Heat Conduction Program," ORNL/CSD/TM-15 (1977).

5. *Nuclear System Materials Handbook,* TID-26666, Hanford Engineering Development Laboratory, Richland, Washington, Vol. 1.

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