Duke Power Company
P.O. Box 1006
Charlotte, NC 28201-1006

M. S. TUCKMAN
Senior Vice President
Nuclear Generation
(704)382-2200 Office
(704)382-4360 Fax

DUKE POWER

September 12, 1995

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: McGuire Nuclear Station
Docket Numbers 50-369 and -370
Catawba Nuclear Station
Docket Numbers 50-413 and -414
Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
Request for Additional Information Relative to DPC-NE-3000P, Revision 1;
Responses to Questions

By letters dated November 15, 1991 (for application to McGuire and Catawba Nuclear Stations) and August 8, 1994 (for application to Oconee Nuclear Station), the NRC transmitted safety evaluations for the subject Topical Report. By letter dated August 9, 1994, Duke Power Company submitted for NRC review Revision 1 to the approved Topical Report. The NRC staff issued a request for additional information (RAI) dated September 6, 1995. Responses to the questions contained in the RAI are presented in Attachment II.

Please note that the responses to several of the questions contain information that Duke considers proprietary. In accordance with 10CFR 2.790, Duke requests that this information be withheld from public disclosure. An affidavit which attests to the proprietary nature of this information is included as Attachment I. Attachment III contains a non-proprietary version of the responses.

APOI

Ltr. Encl.
1 1/NP

Change: NRC PDC

PRIORITY 1

(ACCELERATED RIDS PROCESSING)

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9509190206 DOC.DATE: 95/09/12 NOTARIZED: YES DOCKET #
FACIL:50-269 Oconee Nuclear Station, Unit 1, Duke Power Co. 05000269
50-270 Oconee Nuclear Station, Unit 2, Duke Power Co. 05000270
50-287 Oconee Nuclear Station, Unit 3, Duke Power Co. 05000287
50-413 Catawba Nuclear Station, Unit 1, Duke Power Co. 05000413
50-414 Catawba Nuclear Station, Unit 2, Duke Power Co. 05000414
50-369 William B. McGuire Nuclear Station, Unit 1, Duke Powe 05000369
50-370 William B. McGuire Nuclear Station, Unit 2, Duke Powe 05000370
AUTH.NAME AUTHOR AFFILIATION
TUCKMAN,M.S. Duke Power Co.
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Forwards proprietary responses to request for addl info re Rev 1 to Topical Rept DPC-NE-3000P.Encl withheld (ref 10CFR2.790).

DISTRIBUTION CODE: AP01D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 13+8
TITLE: Proprietary Review Distribution - Pre Operating License & Operating R

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD2-2 LA	1	1	PD2-2 PD	1	1
	PD2-2 PD	1	1	WIENS,L	1	1
	MARTIN,R	1	1	NERSES,V	1	1
INTERNAL:	ACRS	6	6	FILE CENTER 01	1	1
	OGC/HDS2	1	0			
EXTERNAL:	NRC PDR	1	1NP			

P R I O R I T Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM OWFN 5D8 (415-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 15 ENCL ~~13~~ 14

MAF

Duke Power Company
P.O. Box 1006
Charlotte, NC 28201-1006

M. S. TUCKMAN
Senior Vice President
Nuclear Generation
(704)382-2200 Office
(704)382-4360 Fax

DUKE POWER

September 12, 1995

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: McGuire Nuclear Station
Docket Numbers 50-369 and -370
Catawba Nuclear Station
Docket Numbers 50-413 and -414
Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
Request for Additional Information Relative to DPC-NE-3000P, Revision 1;
Responses to Questions

By letters dated November 15, 1991 (for application to McGuire and Catawba Nuclear Stations) and August 8, 1994 (for application to Oconee Nuclear Station), the NRC transmitted safety evaluations for the subject Topical Report. By letter dated August 9, 1994, Duke Power Company submitted for NRC review Revision 1 to the approved Topical Report. The NRC staff issued a request for additional information (RAI) dated September 6, 1995. Responses to the questions contained in the RAI are presented in Attachment II.

Please note that the responses to several of the questions contain information that Duke considers proprietary. In accordance with 10CFR 2.790, Duke requests that this information be withheld from public disclosure. An affidavit which attests to the proprietary nature of this information is included as Attachment I. Attachment III contains a non-proprietary version of the responses.

180051

APOI

Ltr. Encl.
1 1 NP

Change: NRC PDC

If any additional information is needed, please call Scott Gewehr at (704) 382-7581..

Very truly yours,

M. S. Tuckman

M. S. Tuckman

cc (w/ Attachments):

Mr. V. Nerses, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop 14H25, OWFN
Washington, D. C. 20555

L. A. Weins
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop 14H25, OWFN
Washington, D. C. 20555

Mr. R. E. Martin, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop 14H25, OWFN
Washington, D. C. 20555

(w/o Attachments):

Mr. S. D. Ebneter, Regional Administrator
U.S. Nuclear Regulatory Commission - Region II
101 Marietta Street, NW - Suite 2900
Atlanta, Georgia 30323

Mr. G. F. Maxwell
Senior Resident Inspector
McGuire Nuclear Station

Mr. R. J. Freudenberger
Senior Resident Inspector
Catawba Nuclear Station

Mr. P. E. Harmon
Senior Resident Inspector
Oconee Nuclear Station

AFFIDAVIT OF M. S. TUCKMAN

1. I am Senior Vice President, Nuclear Generation Department, Duke Power Company ("Duke"), and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing, and am authorized to apply for its witholding on behalf of Duke.

2 I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission ("NRC") and in conjunction with Duke's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10 CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs, relative to a method of analysis that provides a competitive advantage to Duke.

   (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.790, it is to be received in confidence by the NRC.

   (iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

   (v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the report DPC-NE-3000, "Thermal Hydraulic Transient Analysis Methodology" and supporting documentation, and omitted from the non-proprietary versions.

M. S. Tuckman

M. S. Tuckman

(continued)

This information enables Duke to:

(a) Respond to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions."

(b) Respond to NRC requests for information regarding the transient response of Babcock & Wilcox and Westinghouse pressurized water reactors.

(c) Support license amendment and Technical Specification revision requests for Babcock & Wilcox and Westinghouse PWRs.

(d) Perform safety reviews per 10 CFR 50.59.

(e) Enhance operation of and training programs related to nuclear power plants.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(continued)

AFFIDAVIT OF M. S. TUCKMAN (Page 3)

M. S. Tuckman, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

M. S. Tuckman

M. S. Tuckman

Sworn to and subscribed before me this 12th day of September 1995. Witness my hand and official seal.

Linda Case Smith

Notary Public

My commission expires May 6, 2000

Attachment 3

Question 1

Justify the proposed use of the[ ]for material properties and the[ ]for the fuel gap conductivity by demonstrating that computations will result in conservative system predictions for all transients. (p. 2-40)

Response

This question concerns modeling which has not been revised. Conservative results are obtained as follows. The fuel gap[ ]in order to conservatively model both the stored energy in and heat transfer from the fuel rods to the reactor coolant, and fuel temperature feedback reactivity. This approach results in conservative fuel temperatures and power response, although the material properties of the fuel and cladding are[ ]The fuel temperature is the key modeling parameter, as noted in DPC-NE-3002.

Question 2

Discuss the situations in which[ ]during steady-state initialization may not result in conservative prediction of the transient calculations and reconcile that result with the response to Question 1. (p. 2-40)

Response

In order to obtain steady-state initialization, RETRAN adjusts the SG heat transfer area. If this adjustment is significant, then there will be an impact on the analysis results. To minimize this heat transfer area deviation,[ ]can be employed. This[ ]The heat transfer area reduction cannot be physically justified. Given a choice between these two, the [ ]is preferred. Overall performance of the Oconee RETRAN model has been demonstrated by validation to plant transient data. Adequate conservatism in the heat transfer between the primary and secondary systems is ensured through the assumptions made for the following parameters: Reactor Coolant System flow, steam generator tube plugging, and steam generator level.

Question 3

In the previously submitted model with the original topical report, DPC observed that a large adjustment in the[ ]was necessary during the outsurge portion of any transient containing a strong outsurge. Discuss how this problem is addressed by the use of the revised PZR model which includes modeling of the surge line.

Response

The previously submitted Oconee RETRAN model included modeling of the pressurizer surge line. The surge line metal as a heat conductor was not modeled. The adjustment in the[ ] is related to the solution of the conservation of momentum equation. The heat capacity of the surge line metal is related mainly to the solution of the conservation of energy equation. Therefore, this modeling change is not expected to impact the pressure response during an outsurge.

Question 4

Demonstrate by reanalysis of transients/tests that the revised PZR model with heat conductors results in adequately conservative predictions. In addition, DPC should qualify its PZR water level prediction procedure. (p. 2-42 & p. 3-47)

Response

The revised pressurizer model includes a more detailed treatment of the heat transfer to the pressurizer walls through the use of the [ ] This should be a improvement over the currently approved model in the predictive capabilities of the pressurizer model. As discussed in the DPC response to Question 5 from the NRC letter dated April 7, 1989, a loss of main feedwater at Oconee was actually analyzed employing the improved pressurizer model. The adequacy of the model is discussed in detail in §4.1.1 of the topical report.

The RETRAN pressurizer level control system exactly duplicates the temperature compensated level indication circuit in place at Oconee. The only method to truly validate the level prediction would require knowledge of the actual water level in the pressurizer vessel, which is not available. Discrepancies in the pressurizer level indication observed during plant transient benchmarks are not mainly due to level modeling inaccuracies, but rather are the result of deviations in the predicted reactor power level or primary to secondary heat transfer.

This revised model was benchmarked against plant data from a turbine trip from 37% power. This event is characterized by a reactor power decrease from 37% to 28% over 27 seconds. The pressurizer level response is shown in the attached Figure 4-1. RETRAN slightly overpredicts the pressurizer level during the first 15 seconds. After 15 seconds, RETRAN slightly underpredicts the pressurizer level. These deviations are the result of the integrated effects of the core power and primary-to-secondary heat transfer predictions versus plant data. The overall comparison of the pressurizer level prediction to data is reasonable.

Question 5

Discuss modeling of phase separation including the selected BR velocity in the [ ] (p. 2-49 & 51)

Response

The values for the separation velocity and bubble gradient are the same as those used for the pressurizer volume [ ] and are typical of the RETRAN community. Past experience with the model has shown these values to perform adequately for this application. Particularly since the [ ] is essentially a dead-ended volume, the selection of the bubble rise parameters does not have a significant impact on the transient results. Modeling of a bubble rise volume instead of an HEM volume can have a significant impact on some transients. This model change addresses that limitation.

Question 6

Provide thorough discussion and qualification of the revised SG model for feedring SGs including steady-state initialization and nodalization sensitivities, and demonstrate that the model produces an adequately conservative prediction of heat transfer. In addition, DPC should qualify the SG level calculator for the feedring SG against the data.

Response

The secondary side nodalization for the feedring steam generator has essentially the same level of detail as the preheater-type steam generator model, which has been shown to be adequate. The only significant difference between the two models is that both the lower downcomer and tube bundle regions are [ ] in the feedring steam generator model.

In the absence of a preheater, it was no longer necessary to [ ] The feedring model employs [ ] in the preheater model to more accurately predict primary to secondary heat transfer. These secondary side nodes are coterminous with the primary side tube bundle nodes.

A sensitivity study was performed on the lower downcomer nodalization in an attempt to more closely match vendor data for steam generator level and liquid mass. Based on this sensitivity study, the final nodalization scheme includes [ ]

Obviously there is, as of yet, no plant data to benchmark the feedring steam generator model against. As mentioned above, a good correlation with the manufacturer's calculated data for steam generator level and liquid mass was achieved. The heat transfer prediction was validated based on a comparison of RETRAN results versus vendor data for primary system hot and cold leg temperatures given a specified RCS flow and steam line pressure. Also, the RETRAN feedring steam generator model produces a reasonable void profile over the height of the tube bundle volumes. Both the DPC RETRAN results and the vendor code predictions for the key modeling parameters are shown in the table below:

Parameter	DPC	Vendor
SG outlet pressure (psia)	1020	1020
T_{hot} (°F)	612.75	613.76
T_{cold} (°F)	555.66	556.36
SG Level (%)	65	65
SG Liquid Mass (lbm)	120,124	120,369

Question 7

Discuss the source(s) of the significant reduction in trip setpoints for the load rejection controller for Catawba.

Response

Subsequent to the submittal of the DPC-NE-3002 revision, another modification to the load rejection steam dump setpoints was discovered. The current values are as follows:

Bank	Setpoint (°F)
1	
2	
3	
4	
5	

This is essentially a plant operations issue, since no credit is taken for the non-safety load rejection steam dump controller in any of the Chapter 15 transient analyses. The load rejection controller setpoints have been fine tuned in order to provide better protection in the event of a major load rejection. As a result of the post-modification testing which is to follow the installation of the feedring steam generators, additional adjustments to these setpoints might be made. Revision of these setpoints in the topical report was intended only to keep the document current. Future setpoint changes will be similarly updated as the opportunity arises.

Question 8

Clarify the new paragraph to be inserted in page 3-16 regarding the SG level control. Do both Catawba Units have the DFCS? Discuss how this system is simulated and qualified in the RETRAN analysis.

Response

The original steam generator level control system was replaced by the Digital Feedwater Control System (DFCS) at both Catawba Units. A RETRAN model of the controller has been created, including actual plant values for the controller setpoints, gains and time constants. Prior to implementation in any RETRAN analyses, this model would be validated by benchmarking against plant data taken with the DFCS in place. Currently, however, the steam generator level control system is modeled indirectly as described in §3.2.4.4. This simplified method is used since its only impact on the transient results is in the avoidance of reactor protection and engineered safeguards actuations.

Question 9

Discuss the source(s) and reasons for changes and impact on safety analysis in the following plant models, setpoints and values:

a. HHSI pump characteristics
b. IHSI pump characteristics
c. LHSI pump characteristics

d. Steam line pressure for SI signal & steam line isolation
e. Elimination of a RPS condition for reactor trip
f. Steam line safety valve opening setpoints
g. HHSI and IHSI injection after 7 hours

Response

a, b, & c) Updated vendor information regarding the pump runout limitations on the HHSI and IHSI pumps necessitated modifications to the ECCS flow balancing procedure. These modifications yielded the revised shutoff pressures and runout flows. The values given are consistent with the revised Technical Specifications which were approved on December 15, 1993 for Catawba and July 29, 1994 for McGuire. Revised ECCS injection flow rates have been generated for use in both LOCA and non-LOCA safety analyses.

d) The modification to the low steam line pressure safety injection and steam line isolation Technical Specification setpoints was proposed in the McGuire 1 Cycle 8 reload submittal. The removal of the dynamic compensation of the steam line pressure signal, which accompanies the change in the low pressure setpoint, was intended to eliminate the spurious ESF actuation on minor (but rapid) pressure decreases in the secondary system. The revised steam line pressure setpoint is consistent with all licensing basis safety analyses. The NRC granted the Tech Spec change on November 27, 1991.

e) The removal of the negative flux rate trip from the Technical Specifications was proposed in the McGuire 1 Cycle 8 reload submittal. Based on the elimination of unnecessary reactor trips resulting from mild reactor power transients and the fact that no credit is taken for this trip in the accident analyses, the NRC granted the Tech Spec change on November 27, 1991.

f) The modification to the Bank 4 and 5 SMSV lift setpoints was necessitated by the turbine trip peak secondary pressure analysis. DPC is currently pursuing a change to the way the SMSV lift is modeled which has eliminated the need for these setpoint changes. Therefore, an amended submittal to the NRC will be made shortly.

g) Realignment of the safety injection flow to the hot legs is performed to preclude post-LOCA boron precipitation in the reactor core. Due to increased maximum boron concentrations in the injection water sources (Cold Leg Accumulators, Refueling Water Storage Tank, and the Ice Condenser), the hot leg recirculation switchover time has been changed from 15 to 7 hours.

Question 10

Once the planned steam generator replacement takes place, what does DPC plan to do with respect to the aspects of the report addressing the old SGs for McGuire #1 and 2 and Catawba #1 which would no longer be applicable? Provide comparable benchmark analysis to be included in the topical report in support of the new steam generators.

Response

DPC does not intend to remove any of the discussion or schematics referring to the split flow preheater steam generator designs that are currently in place at both McGuire units and Catawba Unit 1. This is but a small portion of the topical report which, for the most part, discusses McGuire and Catawba generically. This discussion will remain valid for Catawba Unit 2.

As mentioned in the response to Question 6, prior to the installation of the feedring steam generators, there is no transient data available for benchmarking analysis. DPC feels that the differences between the preheater-type steam generators and the feedring steam generators are relatively minor and should not effect the ability of the RETRAN code to accurately predict transient behavior.

Question 11

Clarify Section 3.1.6.2. Which unit at Catawba does the revised AFW runout protection apply to and how is the other unit protected?

Response

The revised AFW runout protection applies to both Catawba units. The basis for the active runout protection which is to be removed is to ensure that a minimum flow requirement is met. This flow requirement is based on a vendor analysis which has been superseded and is no longer part of the DPC licensing basis.

Question 12

Discuss the impact of installation of feedring steam generators and its accompanying changes on transient analysis such as the MFW and AFW flow.

Response

The major design differences in the feedring steam generator with respect to the preheater design include the following: a) the main feedwater enters the annular downcomer of the steam generator through a feedring near the top of the tube bundle as opposed to entering a preheater region at the bottom of the tube bundle, b) the tube bundle itself is about 8 feet taller and contains approximately 2000 more tubes of a slightly smaller diameter, and c) the steam generator liquid mass is approximately 20,000 lbm greater at full power.

Two of the licensing basis analyses where the effects of these design changes are most evident are the feedwater system pipe break and steam generator tube rupture events. The impact of the feedring steam generators on the RETRAN transient analyses for these two events is discussed in the DPC responses to Questions 8 and 12 from the NRC Request for Additional Information regarding the DPC-NE-3002 revision.

Question 13

Discuss in detail how the general transport model is used to simulate boron transport, including the nodalization of injection site, mixing coefficient, analysis for which this is credited, and demonstrate that the model produces conservative results.

Response

This revision does not introduce any changes to the approved boron injection modeling methodology discussed in detail in §5.3.2.5 of DPC-NE-3001. Currently, the general transport model is used only in the steam line break and inadvertent opening of a steam generator relief or safety valve analyses. The injection of the borated safety injection water is modeled at both the high and intermediate head safety injection nozzles. Fill junction boron concentration is determined by control system assuming a mixing coefficient of 1.0.

The conservatisms built into the boron injection modeling include: a conservatively long safety injection response time with no credit taken for flow delivery until the injection pumps have reached full speed, a conservatively high purge volume of unborated water (up to the Refueling Water Storage Tank isolation valves), and a conservatively low Refueling Water Storage Tank boron concentration which includes measurement uncertainties.

Question 14

Discuss the change in assumed steady-state pump head and flow for various transients (§ 2.2.6.2).

Response

The referenced change is editorial in nature. The reactor coolant pump flow rate given in the original topical report was a bounding low value. This is replaced by a best-estimate flow rate which is based on precision calorimetric plant data. A best-estimate value is consistent with the text of the section.

For Chapter 15 accident analyses, the RCS flow rate is adjusted to a bounding value in order to be conservative with respect to the specified acceptance criterion. The pump developed head is adjusted as necessary to be compatible with the desired flow rate assumption.

ONS3 11/14/88 TURBINE TRIP FROM 37% POWER

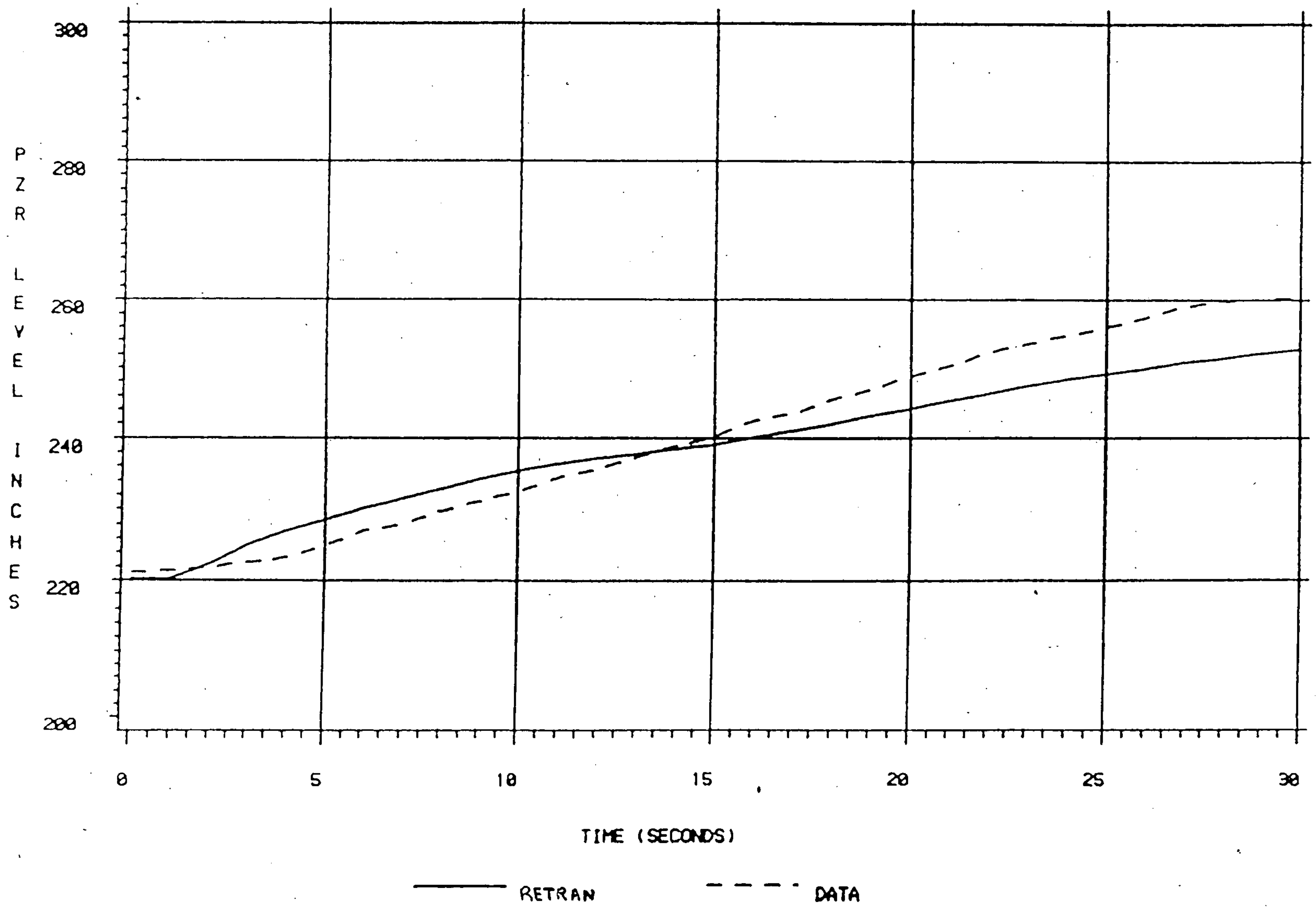

Figure 4-1