The Light company
Houston Lighting & Power South Texas Project Electric Generating Station P. O. Box 289 Wadsworth, Texas 77483

June 26, 1997
ST-HL-AE-5679
File No.: G02.05
10CFR50.54(a)

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project
Unit 1 and Unit 2
Docket No. STN 50-498 and STN 50-499
Response to Request For Additional Information of June 13, 1997
Regarding the South Texas Project's Graded Quality Assurance Program

References:
1) Letter from M. A. McBurnett to the U. S. Nuclear Regulatory Commission dated May 21, 1997, "Revised Graded Quality Assurance Operations Quality Assurance Plan" (ST-HL-AE-5655)
2) Letter from W. T. Cottle to the U. S. Nuclear Regulatory Commission dated May 22, 1997, "Finalized Graded Quality Assurance Operations Quality Assurance Plan" (ST-HL-AE-5661)
3) Letter from L. E. Martin to the U. S. Nuclear Regulatory Commission dated June 10, 1997, "Change QA-032 to the Operations Quality Assurance Plan Revision 13," (ST-HL-AE-5668)
4) Letter from Thomas W. Alexion (NRC) to William T. Cottle, dated June 13, 1997, "Review of Revised Operations Quality Assurance Plan (OQAP), South Texas Project, Units 1 And 2 (STP) (TAC Nos. M92450 And M92451)"

On May 21, 1997, the South Texas Project provided a draft version of the Operations Quality Assurance Plan which implements the Graded Quality Assurance Program for the Nuclear Regulatory Commission review (Reference 1). This version included responses to the requests for additional information provided to the South Texas Project prior to May 21, 1997. Concurrent with the Nuclear Regulatory Commission review, the South Texas Project completed its internal review of the Operations Quality Assurance Plan and on May 22, 1997, the South Texas Project submitted the finalized version (Reference 2).

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On May 29, 1997, the South Texas Project participated in a phone call with the Nuclear Regulatory Commission to discuss NRC comments on the South Texas Project Graded Quality Assurance OQAP. As a result of this phone call, changes were made to the OQAP. These changes were submitted to the Nuclear Regulatory Commission on June 10, 1997 (Reference 3). Your request for additional information dated June 13, 1997, includes the majority of the questions asked during the May 29 phone call. As such, most of your questions have been already answered in Reference 3. Attachment 1 provides responses to the remaining questions which were not previously addressed.

On June 25, 1997, a separate teleconference was held between the South Texas Project and the Nuclear Regulatory Commission, to determine if additional questions existed. As a result of this call, six additional items were requested. The response to these "Request for Additional Information" have been added to Attachment 1 and are clearly identified as such.

The copies of various procedures, reports, contracts and other documents are included in the attached response for information.

If there are any questions regarding this the Operations Quality Assurance Plan, please contact Mr. R. J. Rehkugler at (512) 972-7922. If you have any questions regarding the Graded Quality Assurance Probabilistic Safety Assessment, please contact Mr. C. R. Grantom at (512) 972-7372.

L. E. Martin

L. E. Martin
General Manager,
Nuclear Assurance &
Licensing

JMP/

Attachment:
1) Response To NRC Request for Additional Information of June 10,1997 on the Graded Quality Assurance Program
2) Graded Quality Assurance Process Flowchart
3) Probabilistic Risk Importance Threshold For Input To Graded Quality Assurance Component Classifications
4) Basis for Risk Importance Threshold
5) Additional PSA Information
6) Houston Lighting & Power Audit of PLG , Incorporated Vendor Audit No. 95-073 (VA)
7) STP Nuclear Safety Evaluation Report of PSA Program
8) Documentation of Appendix B Application to PSA Vendor
9) PLG, Inc., Review of STP Interim Model and STP Response

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* Distributed with all Attachments
▸ Distributed with Attachments 1, 2, 3, 4 and 5 only

Response To NRC Request for Additional Information of June 10,1997 on the Graded Quality Assurance Program

Request for Additional Information #1

"Definitions", p. 4 of 10 - The definition of "critical characteristic" needs to be revised to be consistent with the definition given in 10 CFR 21.3.

Response 1

The OQAP definition of "critical characteristics" has been changed. See changes that were submitted under OQAP change 32 (ST-HL-AE-5668).

Request for Additional Information #2

Chapter 1.0, §5.1.4.2, p. 3 of 4 - What are the full responsibilities of the Manager, Risk Management & Industrial Relations?

Response 2

The responsibilities of the Manager, Risk Management and Industrial Relations, as they apply to the Graded Quality Assurance Program, have been included in OQAP change 32 (ST-HL-AE-5668). Other responsibilities are not included, as the South Texas Project does not address personnel responsibilities at this level in the OQAP.

Request for Additional Information #3

*Chapter 2.0, **§3.1,** p. 1 of 15 - "Station economics" should not be a factor in considering the safety needs for a nuclear power plant.*

Response 3

This has been deleted in OQAP change 32 (ST-HL-AE-5668).

Request for Additional Information #4

Chapter 2.0, §2.2, p. 1 of 15 - Pl ase provide explanatory words for including "(except design and fabrication of NRC certified radioactive waste shipping casks)."

Response 4

The exception is currently in place and has been docketed and approved. This exception was taken in September, 1991 (refer to ST-HL-AE-3856) when HL&P clarified that the OQAP (with regard to 10CFR71, Subpart H), applies only to packaging and shipping of radioactive materials and not to design and fabrication of NRC certified radioactive waste shipping casks. HL&P is not imposing design and/or fabrication requirements on casks which have been certified by the NRC. This change (QA-001) was incorporated into the OQAP in December, 1991.

Request for Additional Information #5

*Chapter 2.0, **§5.3.3,** p. **4** of 15 - Add "Initial evaluations are performed by the Working Group." to the end of the paragraph.*

Request for Additional Information #6

*Chapter **2.0. §5.3.5, p. 4** of **15** - After "are" in the first sentence, add "developed by the Working Group and are."*

Request for Additional Information #7

*Chapter 2.0, §5.3.10, p. 5 of 15 - After "experience", add "that could result in recategorization of any SSC." In the next sentence after "are", add "also used." (These suggested changes provide an acceptable response to question **#9** of NRC's 04/14/97 letter).*

Response 5, 6, 7

These changes have been included as part of OQAP change 32 (ST-HL-AE-5668).

Request for Additional Information #8

Chapter 2.0, Note, p. 5 of 15 - It appears that this note is redundant to §5.3.9 above.

Response 8

This note has been removed in OQAP submitted May 22, 1997 (ST-HL-AE-5661).

Request for Additional Information #9

Chapter 2.0, Table I, p. 14 of 15 - For the BASIC program exception to §12 of ANSI N45.2.13-1976, add "for audit of suppliers" after "necessary-"

Request for Additional Information #10

Chapter 13.0, p.4 of 4 - Add a new §5.8 as follows to provide an acceptable response to question #4 of NRC's 04/14/97 letter:

> *"5.8 for medium and low safety significant SSCs treated by the BASIC program controls, measures shall be established to conduct apparent cause determinations and to trend failures to assist in evaluating the need for more detailed root cause analyses (if excessive failures occur) and proper corrective action. Further, particular consideration will be given to assessing the potential implications of such failures generically to similar SSCs treated by the FULL program."*

Response 9, 10

These changes have been included as part of OQAP change 32 (ST-HL-AE-5668).

Request for Additional Information #11

During the May 5-8, 1997, site visit NRC expressed concern that placing components with a risk achievement worth (RAW) greater than 10 but less than 100 in the Basic program may be inappropriate. NRC requested that HL&P identify this population of components in the QA program description, and describe how specific QA controls would be assigned to the components' critical attributes. NRC has not found a satisfactory resolution to this concern in the May 21, 1997, revised submittal. NRC requests that STP change the QA program description to:

- *include a clear definition of the population of components in question. These components are currently categorized as medium safety-significant which provides no distinction from other medium safety-significant populations. NRC is willing to consider the acceptability of a definition of this population which does not include numerical guidelines in the OQAP, but the basic attributes of the population (e.g., high reliability yet a high impact on risk if problems develop) must be clearly described.*

- *provide a description of how QA controls will be assigned to the critical attributes of this population of components. As discussed, NRC does not find that simple application of Basic program controls is sufficient. Nor does NRC find that explicit consideration by the working group and expert panel of the assigned controls is sufficient. NRC is willing to consider the acceptability of assigning Full program controls to those critical component attributes which cause the component to belong to this population.*

Another alternative is to simply assign these components to the high safety-significant category based on the sensitivity of plant risk on their performance and place them in the FULL program. Other alternatives may also be suggested.

Response 11

The South Texas Project has changed the Graded Quality Assurance Program to require safety related components with a RAW between 10 and 100 to have FULL Quality Assurance controls applied to the critical attributes associated with that RAW. The OQAP chapter 2, sections 5.3.9 and 5.3.11 have been revised to reflect this change (OQAP change 32 (ST-HL-AE-5668)).

The Comprehensive Risk Management Procedure, 0PGP02-ZA-0003 Addendum 2 will be revised to incorporate the flowchart provided in Attachment 2 which identifies the Probabilistic Risk Importance thresholds used for Graded Quality Assurance component classifications.

The Graded Quality Assurance Working Group Procedure is currently being developed. It will include the following aspects:

- Components with a risk achievement worth greater than 100 or a Fussell-Vesely importance greater than 0.01 are to be placed in the Full QA Program.
- Components with a risk achievement worth greater than 10 but less than 100 are to have full QA controls specifically placed on those critical attributes which cause the components to have a high risk achievement worth.

A graphical representation of the Probabilistic Risk Importance thresholds for input to the Graded Quality Assurance component classifications is provided in Attachment 3.

Request for Additional Information #12

12. *Although not discussed during the May 5-8, 1997, site visit, discussion among the NRC on the acceptability of your proposed categorization scheme has raise the question of why a high Fussell-Vesely (FV) value should not also lead to a high-safety-significant categorization regardless of the RAW. Please provide your position with respect to this issue.*

Response 12

As noted in the response to item #11, the categorization process has been revised to reflect a threshold for the Fussell-Vesely component importance at 0.01. The basis for risk importance threshold is provided in Attachment 4.

Request for Additional Information #13

13. *Practices and activities to cnsure quality of the South Texas PRA are an important element in justifying use of risk insights as part of the GQA program. It is the staff's understanding that current CDF and LERF values are approximately an order of magnitude lower than in the 1989 (CDF) and 1992 (LERF) baseline studies. Please provide details of processes to ensure that the PRA updates and modifications were correctly implemented. This should include:*

    a listing of the modifications made to the PPA, the reason for each change and a discussion of the impact an the plant's risk profile.

Response 13

The staff is correct in its understanding that the current CDF/LERF values are approximately an order of magnitude lower than the referenced baseline studies. Continuous improvement of South Texas Project's PRA has always been an element of focus. Major PRA applications, such as the recent Diesel Generator Extended Allowed Outage Time (DG EAOT) request, have always contained updated PRA information. Listed below are the major PRA efforts at STP which required model updates along with the associated calculation for CDF and LERF (See also Figure 1 in Attachment 5).

	Core Damage Frequency	Large Early Release Frequency
PRA 1989	1.7×10^{-4} per operating year	Not Calculated
IPE 1992	4.4×10^{-5} per operating year	9.9×10^{-7} per operating year
Tech Spec 1993	3.6×10^{-5} per operating year	1.3×10^{-6} per operating year
DG EAOT 1995	2.1×10^{-5} per operating year	5.6×10^{-7} per operating year
STP_1996	9.1×10^{-6} per operating year	1.4×10^{-7} per operating year

Changes in core damage frequency from the original Probabilistic Safety Assessment (PSA) in 1989 (Reference 1) to the Individual Plant Examination (IPE) in 1992 (Reference 2) are described in IPE Section 1.4.

Changes to the plant models incorporated in the August 1993 submittal to the USNRC for STPEGS Risk-Based Evaluation of Technical Specifications are documented in Reference 3.

The DG Extended Allowed Outage Time (EAOT) study was prepared and submitted to the NRC in April 1995 (Reference 4). This model included enhanced modeling of loss of offsite power, including credit for the emergency transformer and updated offsite power recovery analysis, modeling enhancements based on the On-Line maintenance program at South Texas Project, and the results of the first plant specific data update. The current model was built from the model developed to support the DG EAOT.

No quantification has been made to measure the effect of any single change described below. System level changes were quantified as the system model changes were reviewed and accepted. The quantification of plant model changes were typically made with several changes at once.

Major changes in the current model from the DG EAOT model that affect the Level 1 and Level 2 results include:

- Attempted to obtain the maximum number of cutsets for all systems. Most system models now contain all possible cutsets. The highest cutset cutoff frequency in the current model is 5×10^{-12}. This increased the likelihood of system failure for the affected systems slightly.

- Increased detail in the modeling of planned maintenance of all modeled systems. Slight increase in unavailability for most systems.

- More detailed modeling of all normally operating systems to allow any initial configuration. No change in core damage frequency.

- Development of detailed system specific models for Class 1E 120V AC Vital Power and the Qualified Display Parameter System, Train D Class 1E 125V DC Power, Instrument Air, Solid State Protection System, and Component Cooling Water to the Centrifugal Charging Pumps. Slight increase in core damage frequency as more cutsets could be retained

- Changed the event tree modeling for support systems to represent all possible branches (i.e. 2^n branches where n is the number of top events). This allows more efficient use of logic rules for split fraction assignment. Minor corrections to logic rules were made. Depending upon the specific rule change, an increase or a decrease in core damage frequency resulted, the net effect on plant risk was a slight change.

- Refinement of the Class 1E AC Power model to reflect the bus stripping and breaker closing required after loss of offsite power. This removed these elements from the Emergency Diesel Generator (EDG) models. Slight increase in the likelihood of EDG failure as all system cutsets were obtained. Large increase in the likelihood of failure of the Class 1E AC Power system to reflect the breaker operations necessary to restore power to essential plant equipment.

- Development of split fractions for all systems that reflected all possible operating conditions and boundary conditions of the system. In other words, a three train standby system with one train required for success contains the following system level split fractions:

  - Three Trains Available
  - Trains A and B Available, Train C Failed by Input Conditions
  - Trains A and C Available, Train B Failed by Input Conditions
  - Trains B and C Available, Train A Failed by Input Conditions
  - Train A Available, Trains B and C Failed by Input Conditions
  - Train B Available, Trains A and C Failed by Input Conditions
  - Train C Available, Trains A and B Failed by Input Conditions
  - All Trains Failed by Input Conditions

  In general these changes do not affect core damage frequency or system failure likelihood. These change allow all the basic events in a system to be explicitly included in importance measures.

- Modified the failure distribution for reactor trip breaker mechanical failure to reflect operating information from 1980 to 1993. Decreased the likelihood of ATWS by a factor of 10 with a corresponding change in core damage frequency.

- Ensured consistent modeling of common cause failures in all systems. This increased the likelihood of system failure slightly. No change in most systems.

- Modified the success criteria for the Essential Chilled Water system to include the requirement for cooling the Essential Core Cooling System pump rooms. Slight increase in core damage frequency for LOCA initiators.

- Modified the success criteria for Essential Chilled Water to reflect single train success for general transient events. Slight decrease in core damage frequency.

- Changed the success criteria for the Class 1E 125V DC trains to reflect new station blackout requirements. With a four hour coping time, only one load needs to shed as voltage decreases. No significant change to core damage frequency.

- Either charger in a DC train is capable of supplying all of the DC loads, previously Train A and C DC power required two chargers for success. Slight decrease in system failure frequency.

- Changed the initiating event models for Loss of DC Bus E1A11 or E1B11 to reflect event tree system model. Slight increase in core damage frequency.

- Modified all system specific initiating events to ensure consistent modeling. Changed filter and strainer exposure times to credit the alarms and operator actions specified Alarm Response Procedures. Incorporated the Abnormal Response Procedure for Loss of Ventilation, 0POP04-HE-0001, into the Loss of EAB HVAC and Loss of CR HVAC initiating events. Significant decrease in core damage contribution from these initiators.

The following changes affect the Level 2 models.

- Developed plant specific data on the frequency of opening the Supplemental Purge Valves. The previous data was generated in the mid-1980s based on conversations with operating personnel. The current data is based on plant experience. Reduced the likelihood of Large Early Release.

- Removed the RISKMAN linking event trees and added the necessary information to the Plant Damage State event trees. No significant effect on the Level 2 quantification results.

- Developed a system analysis package for the interfacing systems LOCA analysis. Increased the likelihood of Large Early Release slightly.

- Removed the "Large Pre-existing Leak" failure mode. This failure mode cannot exist if supplemental purge of the containment to reduce containment pressure to comply with Technical Specification requirements is required periodically, as is the case for the STP units. Slight decrease in the Large Early Release frequency.

In addition to the specific changes described above, slight errors in split fraction rule assignment were corrected and minor changes in systems models were incorporated. These changes had no noticeable impact on either core damage frequency or system failure frequency.

With respect to the quality processes performed for the risk model updates the following is a description of the processes used to ensure quality of the STP PRA.

Model revisions to the original PSA in 1989 up to the DG EAOT request were performed by the PRA contractor in conjunction with STP PSA analysts in accordance with the contractor's procedures and guidelines. These changes were reviewed by various groups within STP prior to acceptance and use. The DG EAOT request was issued as a stand alone document and reviewed internally by HL&P. Rather than formal QA procedures, these revisions were prepared and controlled using experienced analysts and peer review to ensure an adequate measure of model control.

The current STP PSA model, STP_1996, was started in September 1995 and was intended from the beginning to be controlled in a manner similar to other processes controlled by quality assurance procedures. This model started from the model used in the analysis of the DG EAOT. The update process was performed to capture changes to the plant (i.e., procedure changes, equipment changes, drawing changes, etc.), correct errors identified during the update process, and to streamline the model to take advantage of the current computer code (RISKMAN®).

The update process was performed by HL&P personnel or by experienced contractor personnel assigned full time to the PRA group. The update was completed in March 1997 and is documented and controlled in a series of system, event tree, and special process notebooks maintained by the PRA group at HL&P. Each of these notebooks was assembled by a designated preparer, reviewed by a person in the PRA group that was not involved in the initial preparation, and accepted by the Risk and Reliability Group Administrator. An interim model was reviewed by the PRA consultant, PLG, who issued a letter report documenting the review. Issues identified by the PRA consultant were resolved and incorporated into the final PRA model. The model is currently undergoing detailed review by Operations and Engineering personnel at STP. The results of these reviews will be incorporated into the next revision to the PSA model.

The update process, although not initially covered by approved quality assurance procedures, was intended to satisfy relevant quality assurance requirements in place for similar processes. The update process correctly identified, modeled, verified, tracked, and implemented revisions to the current PSA.

References pertaining to the response for item #13:

1. South Texas Project Probabilistic Safety Assessment, PLG-0700, prepared for HL&P, April 1989.
2. Level 2 Probabilistic Safety Assessment and Individual Plant Examination, August 1992.
3. HL&P submittal to the U.S. Nuclear Regulatory Commission, "Risk Based Evaluation of Technical Specifications," ST-HL-AE-4544, August 1993.
4. Evaluation of the Proposed Special Test Exception for Diesel Generator and Essential Cooling Water Maintenance, Prepared by HL&P April 1995.

Request for Additional Information #14

During the May 5-8, 1997, site visit, you discussed an audit of your PRA contractors QA program. Please provide the results of the audit or assessment of the QA program of your PRA contractor.

Response 14

Attachment 6 provides that latest Procurement Quality Audit Report 95-073 (VA) of PLG, Incorporated, performed at the PLG's Newport Beach facility in California, on September 11 through 14, 1995. It should be noted that STP owns, controls, and maintains all STP risk models. Contractor organizations are used for staff augmentation or to perform special projects and are not used to maintain or otherwise control the content of STP risk models.

Request for Additional Information #15

In your response to RAI G-1 under cover letter dated October 30, 1996, you wrote that, "recently program procedures were developed to implement Appendix B features to establish configuration control of the PSA models.' We note that we have received four procedures by letter dated May 22, 1997. The May 22, 1997, cover letter also stated that the "Configuration Control of the Probabilistic Safety Assessment Procedure" has been deleted. Please provide us with the procedures which will implement Appendix B features to establish configuration control-of the PRA models, or identify which of the four procedures is intended to provide that control.

Response 15

The requirement for PSA configuration control is contained in the Probabilistic Safety Assessment Program procedure, 0PGP04-ZA-0604, step 5.3 (See Attachment 5). The process used to describe the activities used to maintain configuration control of the PSA is contained in Risk Assessment Guideline 002, Review and Documentation of PSA Input Document Changes (See Attachment 5). The need to reference the PSA configuration control guidance document in the Probabilistic Safety Assessment Program procedure, 0PGP04-ZA-0604 has been determined to be necessary to ensure that changes to the PSA configuration control process are appropriately controlled. The Probabilistic Safety Assessment Program procedure, 0PGP04-ZA-0604, step 5.3 will be revised to reference Risk Assessment Guideline 002 and, in step 5.5 to require that changes to that Risk Assessment Guideline be peer reviewed (See marked up procedure in Attachment 5).

The following "Request for Additional Information" resulted from a teleconference between the South Texas Project and the Nuclear Regulatory Commission on June 25, 1997:

TeleCon Request for Additional Information 1

Need to define in appropriate procedure what "not risk significant" means.

Response to TeleCon Request for Additional Information 1

"Non-Risk Significant - components which are truncated/not modeled by the PSA and with no or negligible deterministic safety importance."

TeleCon Request for Additional Information 2

The definition of "Targeted" in the Comprehensive Risk Management Procedure does not match what is proposed in the Operations Quality Assurance Plan. What will be the QA controls put in place for the critical attributes of non-safety related SSCs put into "Targeted"?

Response to TeleCon Request for Additional Information 2

"Targeted", as it applies to non-safety related SSCs, will be described in the OQAP and appropriate procedure as follows:

> "Targeted" program controls are applied to non-safety related SSCs, for which 10 CFR50 Appendix B is not applicable, categorized as "high" or "medium" safety significant/risk importance. Specific program controls consistent with applicable portions of the full and basic program controls are applied to those items in a selected manner, "targeted" at those characteristics or critical attributes that render the SSC significant or important.

TeleCon Request for Additional Information 3

Have there been any internal QA audits of the Probabilistic Safety Assessment program at the South Texas Project? If so, provide documentation.

Response to TeleCon Request for Additional Information 3

On June 20, 1996, the South Texas Project completed a Nuclear Safety Evaluation of the Probabilistic Risk Analysis program. A copy of the Nuclear Safety Evaluation Report (NSE 96-02) is provided in Attachment 7.

TeleCon Request for Additional Information 4

Is Appendix B imposed on your Probabilistic Safety Assessment vendor? If so, provide purchase order documentation

Response to TeleCon Request for Additional Information 4

Activities performed by PLG, Inc., one of our PSA vendors, is governed by Appendix B. A copy of a purchase order for contract services with PLG, Inc. is provided in Attachment 8.

TeleCon Request for Additional Information 5

PLG issued a letter report, refer to RAI 13 of June 19 telefax documenting its review of your interim model (prior to March 97 update), please send this letter report and the issue resolution.

Response to TeleCon Request for Additional Information 5

A copy of the PLG report along with our response is provided in Attachment 9.

TeleCon Request for Additional Information 6

Provide road map of previously submitted Operation Quality Assurance Plan changes which have been submitted (relative revision designation).

Response to TeleCon Request for Additional Information 6

The biennial update was submitted in December, 1996 (ST-HL-AE-5524, December, 17, 1996). This was revision 12 and incorporated change notices QA-024, 25, 26, 27, 29, and 30. This is noted in Attachment 1 of the noted correspondence.

Change notice QA-028 was used to submit the original graded QA submittal and has been maintained as "pending". All changes concerning graded quality assurance have been made and are tracked as change QA-028. The revised graded quality assurance OQAP, revision 13 is being tracked under QA-028 (ST-HL-AE-5661, 5/22/1997). See description of changes, Attachment 2 to the noted correspondence.

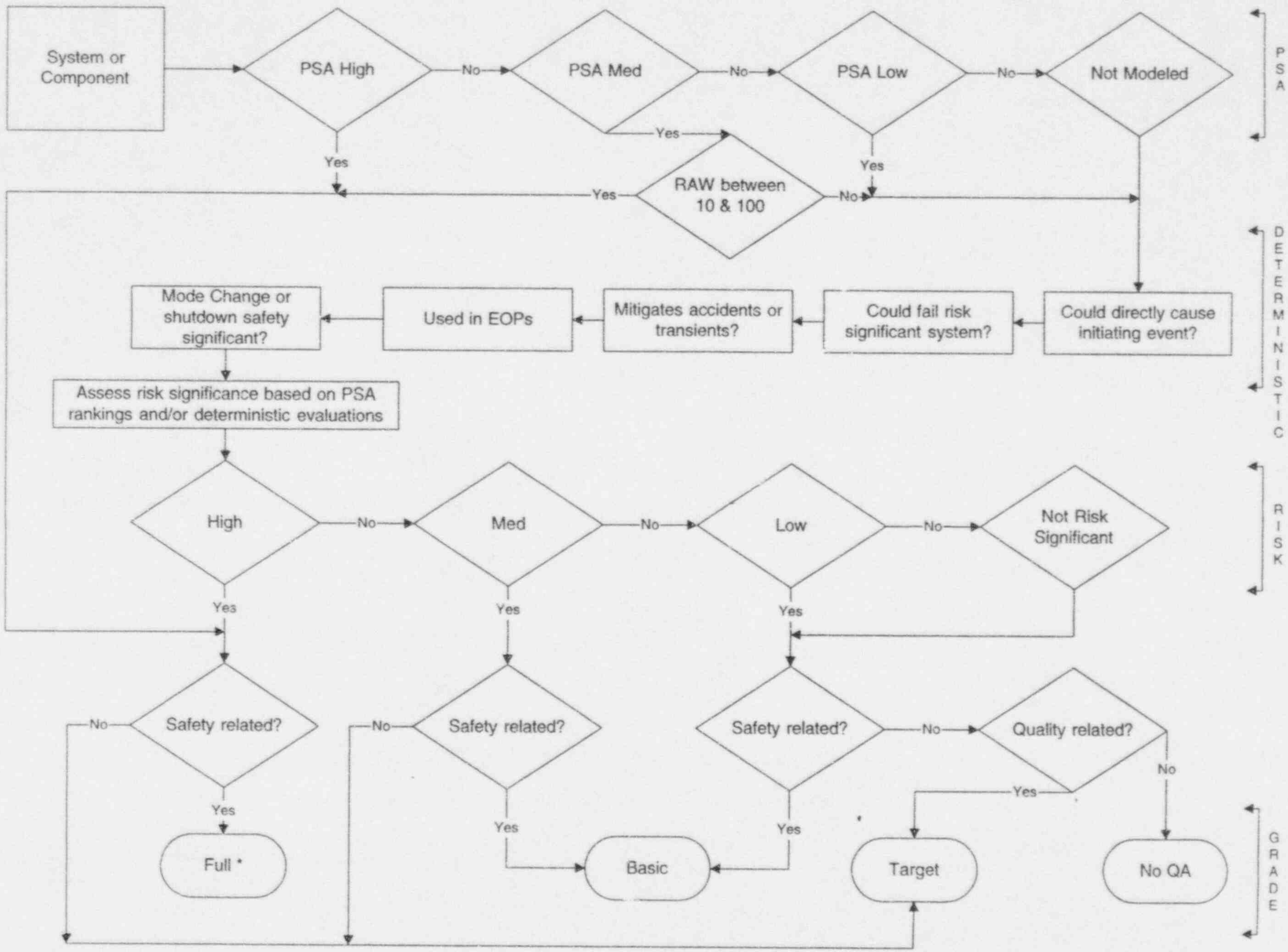
Change notice QA-031 was prepared for revision 12 of our current, approved, effective OQAP and provided an update to the organization and changed the definitions for "commercial grade item" and "dedication." The changes were incorporated into the graded OQAP submitted 5/22/1997 in order to keep that document current.

Change notice QA-032 was prepared for the graded OQAP, revision 13, in response to the RAI. Our administrative controls for the OQAP require changes made between biennial updates to be processed as change notices. Because revision 13 was processed through our normal review and comment process and approved by the Executive Vice President and General Manager, Nuclear, the changes resulting from the RAI were required to be processed as a change notice. The next sequential number was 032.

ATTACHMENT 2

GRADED QUALITY ASSURANCE PROCESS FLOWCHART

GQA PROCESS

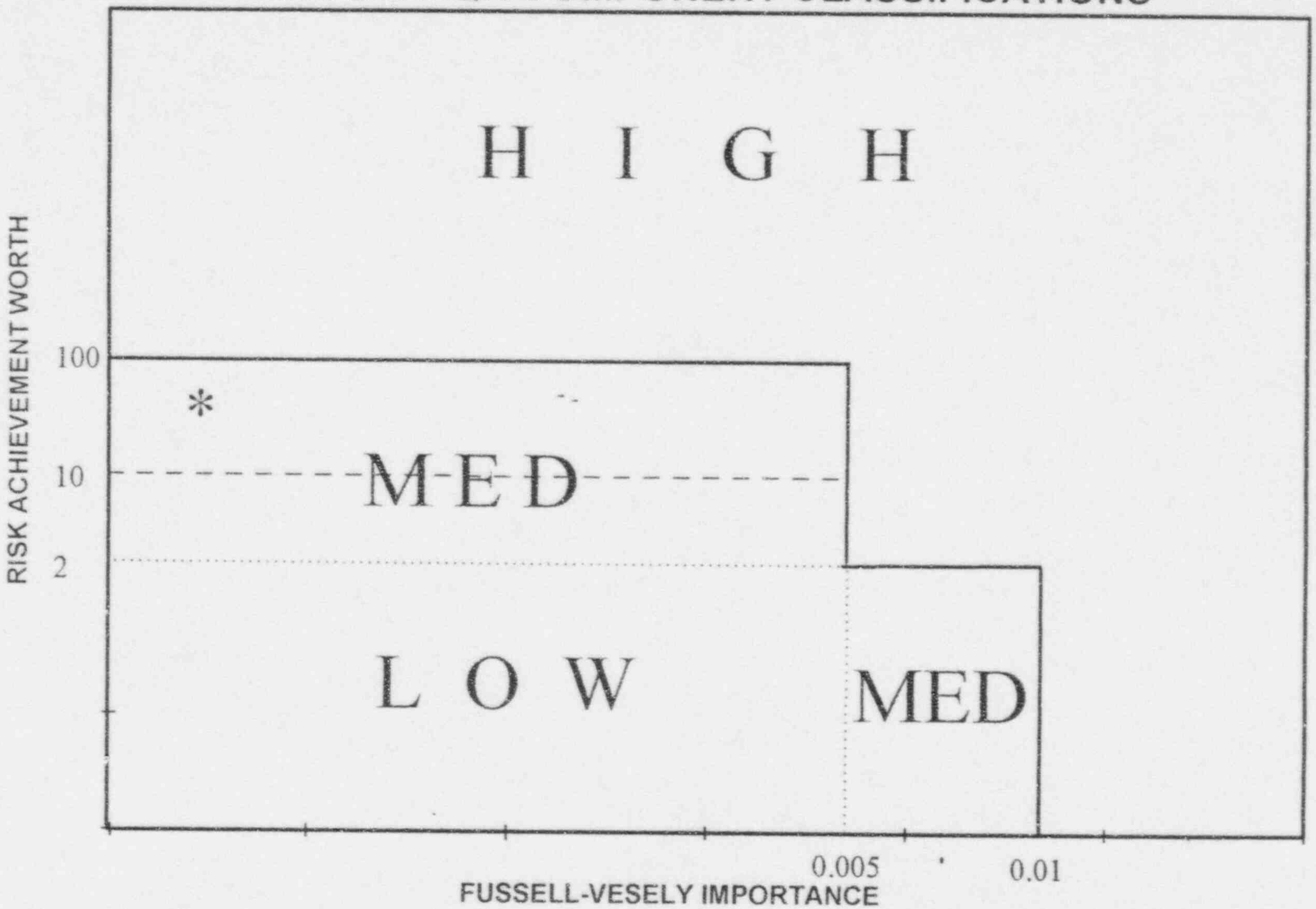


* SR Components with a RAW between 10 and 100 are to have Full QA applied to the critical attributes associated with that RAW

ATTACHMENT 3

PROBABILISTIC RISK IMPORTANCE THRESHOLD FOR INPUT TO GRADED QUALITY ASSURANCE COMPONENT CLASSIFICATIONS

PROBABILISTIC RISK IMPORTANCE THRESHOLDS FOR INPUT TO GRADED QA COMPONENT CLASSIFICATIONS



* Full Program is applied to critical attributes associated with the high Risk Achievement Worth

Ref: STP-1996
Date: 6/3/97
File: Ranking6.ppt